TMI

TABLE OF CONTENTS

Section	Title	Page
I	OPERATIONAL SAFETY	I.A-1
I.A	OPERATING PERSONNEL	I.A-1
I.A.1	OPERATING PERSONNEL AND STAFFING	I.A-1
I.A.1.1	Shift Technical Advisor	I.A-1
I.A.1.2	Shift Supervisor Administrative Duties	I.A-2
I.A.1.3	Shift Manning	I.A-2
I.A.2	TRAINING AND QUALIFICATION OF OPERATING PERSONNEL	I.A-3
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	I.A-3
I.A.2.3	Administration of Training Programs	I.A-4
I.A.3	LICENSING AND REQUALIFICATION OF OPERATING PERSONNEL	I.A-4
I.A.3.1	Revise Scope and Criteria for Licensing Examinations	I.A-5
I.B	SUPPORT PERSONNEL	I.B-1
I.B.1	MANAGEMENT FOR OPERATIONS	I.B-1
I.B.1.2	Evaluation of Organization and Management Improvements of Near – Term Operating License Applicants	I.B-1
I.B.2	INSPECTION OF OPERATING REACTORS	I.B-2
I.B.2.2	Resident Inspector at Operating Reactors	I.B-2
I.C	OPERATING PROCEDURES	I.C-1
I.C.1	SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURES REVISION	I.C-1
I.C.2	SHIFT AND RELIEF TURNOVER PROCEDURES	I.C-2
I.C.3	SHIFT SUPERVISOR’S RESPONSIBILITIES	I.C-3
I.C.4	CONTROL ROOM ACCESS	I.C-3
I.C.5	PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF	I.C-4
I.C.6	PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES	I.C-5
I.C.7	NSSS VENDOR REVIEW OF PROCEDURES	I.C-6
I.C.8	PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL APPLICANTS	I.C-6
I.D	CONTROL ROOM DESIGN	I.D-1
I.D.1	CONTROL ROOM DESIGN REVIEW	I.D-1
I.D.2	PLANT SAFETY PARAMETER DISPLAY CONSOLE	I.D-2
I.G	PREOPERATIONAL AND LOW POWER TESTING	I.G-1
I.G.1	TRAINING REQUIREMENTS	I.G-1
II	SITING AND DESIGN	II.B-1
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW	II.B-1
II.B.1	REACTOR COOLANT SYSTEM VENTS	II.B-1

TABLE OF CONTENTS (Continued)

Section	Title	Page
II.B.2	PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POSTACCIDENT OPERATION	II.B-2
II.B.2.1	Source Terms	II.B-3
II.B.2.2	System Review	II.B-3
II.B.2.2.1	Containment	II.B-3
II.B.2.2.2	Emergency Core Cooling System	II.B-3
II.B.2.2.3	Residual Heat Removal System	II.B-4
II.B.2.2.4	Containment Spray System	II.B-4
II.B.2.2.5	Chemical And Volume Control System	II.B-4
II.B.2.2.6	Gas Waste Processing System	II.B-4
II.B.2.2.7	Post-Accident Sampling System	II.B-4
II.B.2.3	Shielding Methods	II.B-4
II.B.2.4	Design Review	II.B-5
II.B.2.5	Radiation Qualification of Class 1E Equipment	II.B-5
II.B.3	POST-ACCIDENT SAMPLING	II.B-5
II.B.3.1	General Description	II.B-6
II.B.3.2	Design Criteria and Functional Requirements	II.B-6
II.B.3.3	PASS Equipment	II.B-6
II.B.3.3.1	Reactor Coolant PASS	II.B-6
II.B.3.3.2	Containment Air PASS	II.B-7
II.B.3.3.3	PASS Isolation Valve Control	II.B-7
II.B.4	TRAINING FOR MITIGATING CORE DAMAGE	II.B-8
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES	II.D-1
II.D.1	TESTING REQUIREMENTS	II.D-1
II.D.3	RELIEF AND SAFETY VALVE POSITION INDICATION	II.D-3
II.E	SYSTEM DESIGN	II.E-1
II.E.1	AUXILIARY FEEDWATER SYSTEM	II.E-1
II.E.1.1	Auxiliary Feedwater System Evaluation	II.E-1
II.E.1.2	Auxiliary Feedwater System Automatic Initiation And Flow Indication	II.E-10
II.E.3	DECAY HEAT REMOVAL	II.E-14
II.E.3.1	Reliability of Power Supplies for Natural Circulation	II.E-15
II.E.4	CONTAINMENT DESIGN	II.E-16
II.E.4.1	Dedicated Penetration	II.E-16
II.E.4.2	Isolation Dependability	II.E-16
II.F	INSTRUMENTATION AND CONTROLS	II.F-1
II.F.1	ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION	II.F-1
II.F.2	IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING	II.F-3
II.F.2.1	RCS Saturation Margin	II.F-4
II.F.2.2	Collapsed Water Level	II.F-4
II.F.2.3	Reactor Coolant System Temperatures	II.F-5
II.F.2.4	Implementation	II.F-5
II.G	ELECTRICAL POWER	II.G-1
II.G.1	POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES AND LEVEL INDICATORS	II.G-1

TABLE OF CONTENTS (Continued)

Section	Title	Page
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENT AND LOSS-OF-FEEDWATER ACCIDENTS	II.K-1
II.K.1	IE BULLETINS	II.K-1
II.K.2.13	Thermal Mechanical Report – Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident With No Auxiliary Feedwater	II.K-2
II.K.2.17	Potential for Voiding in the Reactor Coolant System During Transients	II.K-3
II.K.2.19	Sequential Auxiliary Feedwater Flow Analysis	II.K-4
II.K.3	FINAL RECOMMENDATIONS FOR B&O TASK FORCE	II.K-4
II.K.3.1	Installation and Testing of Automatic Power-Operated Relief Valve Isolation System	II.K-4
II.K.3.3	Reporting of SV & RV Failures & Challenges	II.K-5
II.K.3.5	Automatic Trip of Reactor Coolant Pumps During Loss-Of-Coolant Accident	II.K-6
II.K.3.9	Proportional Integral Derivative Controller Modification	II.K-6
II.K.3.10	Proposed Anticipatory Trip Modification	II.K-7
II.K.3.11	Justification of Use of Certain PORVs	II.K-7
II.K.3.12	Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip	II.K-8
II.K.3.17	Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes	II.K-8
II.K.3.25	Effect of Loss of Alternating-Current Power on Pump Seals	II.K-8
II.K.3.30	Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance With 10 CFR Part 50, Appendix K	II.K-9
II.K.3.31	Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46	II.K-10
III	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS	III.A-1
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS	III.A-1
III.A.1	IMPROVE LICENSEE EMERGENCY PREPAREDNESS - SHORT TERM	III.A-1
III.A.1.1	Upgrade Emergency Preparedness	III.A-1
III.A.1.2	Upgrade Licensee Emergency Support Facilities	III.A-2
III.A.2	IMPROVING LICENSEE EMERGENCY PREPAREDNESS-- LONG TERM	III.A-11
III.A.3	IMPROVING NRC EMERGENCY PREPAREDNESS	III.A-12
III.A.3.3	Communications	III.A-12
III.D	RADIATION PROTECTION	III.D-1
III.D.1	RADIATION SOURCE CONTROL	III.D-1
III.D.1.1	Primary Coolant Sources Outside The Containment Structure	III.D-1
III.D.3	WORKER RADIATION PROTECTION IMPROVEMENT	III.D-3
III.D.3.3	Improved Inplant Iodine Instrumentation Under Accident Conditions	III.D-3
III.D.3.4	Control Room Habitability Requirements	III.D-4

LIST OF TABLES

Number	Title
II.B.2-1	DELETED
II.B.2-2	DELETED
II.B.2-3	DELETED
II.B.2-4	VITAL AREA ACCESS
II.B.2-5	DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE
II.B.2-6	DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE
II.B.3-1	DELETED
II.D.1-1	SAFETY VALVE INFORMATION
II.E.1.1-1	PRIMARY EVENT PROBABILITIES
II.E.1.1-2	CRITERIA FOR AUXILIARY FEEDWATER SYSTEM DESIGN BASIS CONDITIONS
II.E.1.1-3	SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES
II.E.1.1-4	SUMMARY OF SENSIBLE HEAT SOURCES
II.E.1.1-5	DELETED
II.F.2-1	SATURATION MARGIN MONITOR AND CORE EXIT THERMOCOUPLE SPECIFICATION SUMMARY

LIST OF FIGURES

Number	Title
II.B.2-1	Deleted
thru	
II.B.2-24	
II.B.2-25	Post-LOCA Access Route 1A
II.B.2-26	Post-LOCA Access Route 1B
II.B.2-27	Post-LOCA Access Route 1C
II.B.2-28	Deleted
II.B.2-29	Deleted
II.B.2-30	Post-LOCA Access Route 2
II.B.2-31	Post-LOCA Access Route 3 and 4
II.B.2-32	Post-LOCA Access Route 5
II.B.2-33	Post-LOCA Access Route 6
II.B.2-34	Post-LOCA Access Route 7 and 8
II.B.2-35	Post-LOCA Access Route 9
II.B.2-36	Post-LOCA Access Route 10 and 11 (2 Sheets)
II.B.2-37	Post-LOCA Access Route 12
II.B.2-38	Deleted
II.B.2-39	Post-LOCA Access Route 14 (3 Sheets)
II.B.2-40	Post-LOCA Access Route 15 (2 Sheets)
II.B.2-41	Post-LOCA Access Route 16
II.B.2-42	Post-LOCA Access Route 17
II.B.2-42.1	Post-LOCA Access Route 19
II.B.2-42.2	Post-LOCA Access Route 19

LIST OF FIGURES (Continued)

Section	Title	Page
II.B.2-42.3	Post-LOCA Access Route 19	
II.B.2-42.4	Post-LOCA Access Route 20	
II.B.2-43	Post-LOCA Access Route 1A	
II.B.2-44	Post-LOCA Access Route 1A	
II.B.2-45	Post-LOCA Access Route 1B	
II.B.2-46	Post-LOCA Access Route 1B	
II.B.2-47	Post-LOCA Access Route 1C	
II.B.2-48	Deleted	
II.B.2-49	Deleted	
II.B.2-50	Post-LOCA Access Route 2	
II.B.2-51	Post-LOCA Access Route 2	
II.B.2-52	Post-LOCA Access Route 3 and 4	
II.B.2-53	Post-LOCA Access Route 3 and 4	
II.B.2-54	Post-LOCA Access Route 5	
II.B.2-55	Post-LOCA Access Route 5	
II.B.2-56	Post-LOCA Access Route 6	
II.B.2-57	Post-LOCA Access Route 6	
II.B.2-58	Post-LOCA Access Route 7 and 8	
II.B.2-59	Post-LOCA Access Route 7 and 8	
II.B.2-60	Post-LOCA Access Route 9	
II.B.2-61	Post-LOCA Access Route 9	
II.B.2-62	Post-LOCA Access Route 10 and 11	
II.B.2-63	Post-LOCA Access Route 10 and 11	

LIST OF FIGURES (Continued)

Section	Title	Page
II.B.2-64	Post-LOCA Access Route 10 and 11	
II.B.2-65	Post-LOCA Access Route 12	
II.B.2-66	Deleted	
II.B.2-67	Deleted	
II.B.2-68	Post-LOCA Access Route 14	
II.B.2-69	Post-LOCA Access Route 14	
II.B.2-70	Post-LOCA Access Route 14	
II.B.2-71	Post-LOCA Access Route 14	
II.B.2.72	Post-LOCA Access Route 14	
II.B.2.73	Post-LOCA Access Route 14	
II.B.2-74	Post-LOCA Access Route 15A	
II.B.2-75	Post-LOCA Access Route 15B	
II.B.2-76	Post-LOCA Access Route 15B	
II.B.2-77	Post-LOCA Access Route 15B	
II.B.2-78	Post-LOCA Access Route 16	
II.B.2-79	Post-LOCA Access Route 17	
II.B.2-80	Post-LOCA Access Route 17	
II.B.2-81	Post-LOCA Access Route 19	
II.B.2-82	Post-LOCA Access Route 19	
II.B.2-83	Post-LOCA Access Route 19	
II.B.2-84	Post-LOCA Access Route 19	
II.B.2-85	Post-LOCA Access Route 20	
II.B.2-86	Post-LOCA Access Route 20	

LIST OF FIGURES (Continued)

Section	Title	Page
II.B.3-1	Flow Diagram - Post-Accident Sampling System Reactor Coolant	
II.B.3-2	Containment Ventilation and CA PASS Flow Diagram	
II.E.1.1-1	Auxiliary Feedwater System Simplified Flow Diagram	
II.E.1.1-2	AFWS Simplified Fault Tree, Loss of Feedwater (6 Sheets)	
III.A.1.2-1	Technical Support Center	
III.A.1.2-2	Integrated ERF Computer System	
III.A.1.2-3	Operational Support Center	
III.A.1.2-4	Emergency Operations Facility	

I OPERATIONAL SAFETY

I.A OPERATING PERSONNEL

I.A.1 OPERATING PERSONNEL AND STAFFING

OBJECTIVE:

"Complex transients in nuclear power plants place high demands on the operators in the control room. The objective of the actions described in this task is to increase the capability of the shift crews in the control room to operate the facility in a safe and competent manner by assuring that a proper number of individuals with the proper qualifications and fitness are on shift at all times."

-NUREG 0660, pg. I.A.1-1

I.A.1.1 Shift Technical Advisor

Action Plan Requirements:

"Provide on shift at each nuclear power plant a qualified person (the shift technical advisor) with a bachelor's degree or equivalent in a science or engineering discipline with specific training in the plant response to off-normal events and in accident analysis of the plant."

"Shift technical advisors shall serve in an advisory capacity to shift supervisors. The licensee shall assign normal duties to the shift technical advisor that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience."

-NUREG 0578, pg. 13

Also see NUREG 0737

CPNPP Response

Refer to revised Section 13.2.1.1, Item 5 .

I.A.1.2 Shift Supervisor Administrative Duties

Action Plan Requirements:

"The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room."

- NUREG 0578, pg. A-48

CPNPP Response

A list of administrative duties of the Shift Supervisor will be provided to the Vice President, Nuclear, for his review prior to fuel load.

I.A.1.3 Shift Manning

Action Plan Requirements:

"The minimum shift crew for a unit shall include three operators, plus an additional three operators when the unit is operating. Shift staffing may be adjusted at multi-unit stations to allow credit for operators holding licenses on more than one unit."

"In each control room, including common control rooms for multiple units, there shall be at all times a licensed reactor operator for each reactor loaded with fuel and a senior reactor operator licensed for each reactor that is operating. There shall also be onsite at all times, an additional relief operator licensed for each reactor, a licensed senior reactor operator who is designated as shift supervisor, and any other licensed senior reactor operators required so that their total number is at least one more than the number of control rooms from which a reactor is being operated."

"Administrative procedures shall be established to limit maximum work hours of all personnel performing a safety-related function to no more than 12 hours of continuous duty with at least 12 hours between work periods, no more than 72 hours in any 7 day period, and no more than 14 consecutive days of work without at least 2 consecutive days off."

-NUREG 0694, pg. 11

CPNPP Response

Procedures have been established to address shift manning and an overtime policy. These procedures will be implemented prior to fuel load.

CPNPP currently follows shift manning and overtime policy prescribed by 10 CFR Part 26 Subpart I which supercedes NUREG-0694 criteria.

I.A.2 TRAINING AND QUALIFICATION OF OPERATING PERSONNEL

OBJECTIVE:

“Improve the capability of operators and supervisors to understand and control complex reactor transients and accidents, and improve the general capability of an operations organization to respond rapidly and effectively to upset conditions. Increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capability to perform their duties.”

-NUREG 0660, pg. I.A.2-1

I.A.2.1 Immediate Upgrading of Operator and Senior Operator Training and Qualifications

Action Plan Requirements:

“A. Eligibility Requirements to be Administered an Examination.

“1. Experience*

a. Applicants for senior operator licenses shall have 4 years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of 2 years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least 6 months of the nuclear power plant experience shall be at the plant for which he seeks a license.

b. Applicants for senior operator licenses shall have held an operator’s license for 1 year.”

“2. Training

a. Senior operator*: Applicants shall have 3 months of shift training as an extra man on shift.

b. Control room operator*: Applicants shall have 3 months training on shift as an extra person in the control room.

---

*Precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility has not yet been in operation.

c. Training programs shall be modified, as necessary, to provide:

1) Training in heat transfer, fluid flow and thermodynamics.

2) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

3) Increased emphasis on reactor and plant transients."

"3. Facility Certifications

Certifications completed pursuant to Sections 55.10(a) (6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be signed by the highest level of corporate management for plant operation (for example, Vice President for Operations)."

-NRC Letter dated March 28, 1980,
Enclosure 1

Also see NUREG 0737

CPNPP Response

Refer to revised Section 13.2.1.1. The referenced program is consistent with 10 CFR 55 rulemaking issued after the TMI-2 event.

I.A.2.3 Administration of Training Programs

Action Plan Requirements

"Training center and facility instructors who teach systems, integrated responses, transient and simulator courses shall demonstrate their competence to NRC by successful completion of a senior operator examination."

"Instructors shall be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations."

-NRC Letter dated March 28, 1980,
Enclosure 1

CPNPP Response

Refer to revised Section 13.2.1. The referenced program is consistent with 10 CFR 55 rulemaking issued after the TMI-2 event.

I.A.3 LICENSING AND REQUALIFICATION OF OPERATING PERSONNEL

OBJECTIVE:

"Upgrade the requirements and procedures for nuclear power plants operator and supervisor licensing to assure that safe and competent operators and senior operators are in charge of the

day-to-day operation of nuclear power plants. Increase the requirements for initial issuance of licenses and for license renewals and provide closer NRC monitoring of licensed activities."

-NUREG 0660, Pg. I.A.3-1

I.A.3.1 Revise Scope and Criteria for Licensing Examinations

Action Plan Requirements:

"B. NRC Examinations

1. Increase Scope of Examinations

a. A new category shall be added to the operator written examination entitled, "Principles of Heat Transfer and Fluid Mechanics."

b. A new category shall be added to the senior operator written examination entitled, "Theory of Fluids and Thermodynamics."

c. Time limites shall be imposed for completion of the written examinations:

1. Operator: 9 hours.

2. Senior Operator: 7 hours.

d. The passing grade for the written examination shall be 80% overall and 70% in each category.

e. All applicants for senior operator licenses shall be required to be administered an operating test as well as the written examination.

f. Applicants will grant permission to NRC to inform their facility management regarding the results of the examinations for purposes of enrollment in requalification programs."

-NRC Letter, dated March 28, 1980
Enclosure 1, pg. 4

2. Simulator Examinations

"Simulator examinations will be included as part of the licensee examinations."

"The administration of simulator examinations will be deferred for applicants whose facilities do not have simulators on site as of October 1, 1980. These deferred simulator examinations will be initiated by October 1, 1981."

- NUREG 0737

CPNPP Response

Refer to revised Section 13.2.1.1 . The referenced program is consistent with 10 CFR 55 rulemaking issued after the TMI-2 event.

All CPNPP senior operator cold license candidates will be prepared for an operating test as well as a written examination.

NRC will be provided with a signed release granting permission for NRC to inform CPNPP management regarding the results of the examinations for purposes of enrollment in requalification programs.

CPNPP takes the position that operations on the specified plant should be reinforced by examination at the time of licensing rather than operations on a non-identical plant. Reasoning from this position, CPNPP maintains that simulator examinations on a non-identical plant simulator should not be scheduled consecutively with the balance of the license examination. Furthermore, scheduling examinations at a remote location for an entire operating staff on a fully-booked simulator where the examination target date is not completely predictable cannot be accomplished by a cold license applicant in the narrow window of time prescribed.

I.B SUPPORT PERSONNEL

I.B.1 MANAGEMENT FOR OPERATIONS

OBJECTIVE:

"Improve licensee safety performance and ability to respond to accidents by upgrading the licensee groups responsible for radiation protection and plant operation. The areas to be upgraded include (1) staff size; (2) education and experience of staff members; (3) plant operating and emergency prodecures; (4) management awareness of and attention to safety matters; and (5) numbers and types of personnel available to respond to accidents. Licensee safety performance would further improved if (1) a full-time, dedicated, onsite safety engineering staff were established, and (2) and integrated program for the systematic review of operating experience were provided with the concurrent dissemination of information to plant personnel."

- NUREG 0660, pg. I.B.1-1

I.B.1.2 Evaluation of Organization and Management Improvements of Near – Term Operating License Applicants

Action Plan Requirements:

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The pricipal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent reviews and audits of plant activities including maintenance, modifications, operating problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another functions of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

- NUREG 0737

CPNPP Response

The principal functions and composition of the former ISEG have been distributed throughout CPNPP organizations.

Since the time of NUREG-0737 (1980) and CPNPP licensing and initial operations (1990), the CPNPP organization has continually evolved and improved. Many new plant programs and

processes have been developed and implemented (e.g., improved Corrective Action Program, System Engineering Department and related programs, a Maintenance Rule Program, individual plant department self-assessments, etc.). These improved organizational attributes overlap and/or complement the original ISEG intended functions. Therefore it was determined that continued description of the ISEG in the FSAR was unnecessary.

I.B.2 INSPECTION OF OPERATING REACTORS

I.B.2.2 Resident Inspector at Operating Reactors

"An NRC resident inspector will be assigned to each site."

- NUREG 0694, Pg. 26

CPNPP Response

CPNPP will provide facilities for each resident inspector as required.

I.C OPERATING PROCEDURES

OBJECTIVE:

"Improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance will be included. A principal part of this work is to improve procedures for dealing with abnormal conditions and emergencies by improving the delineation of symbols, events, and plant conditions that identify emergency or off-normal situations that confront the operators and, once identified, to assure consistency with operator training."

- NUREG-0660, pg. I.C-1

I.C.1 SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURES REVISION

Action Plan Requirements:

"a. Provide the analysis, emergency procedures, and training to substantially improve operator performance during a small break loss-of-coolant accident."

"b. Provide the analysis, emergency procedure, and training needed to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling."

"c. Provide the analysis, emergency procedures, and training to substantially improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions."

- NUREG 0578, Pg. 12

Also see NUREG 0737.

CPNPP Response

Small break loss-of-coolant accident analyses have been performed and submitted to Staff in WCAP-9600. The report presents a comprehensive study of Westinghouse system response to small breaks. On-going efforts aimed at improving emergency operating procedure guidelines have already been discussed by Westinghouse and NRC Staff.

Inadequate core cooling is an item to be addressed, as stated in I.E. Bulletin 79-06C. Further definition of the scope of the item is needed form the NRC Staff, such as system failure and operator error assumptions to be made in the analyses. At present Westinghouse model preparation is in progress to permit response to identified action.

Westinghouse does plan to perform pre-test calculations of the LOFT tests when the necessary input information is available.

The purpose of the recommended transient and accident analysis is to provide an increase in safety by improving the performance of reactor operators during transient and accident

conditions. The primary concern is that the operator training and emergency operating procedures are based on the conservative FSAR Chapter 15 analyses. Chapter 15 should continue to be used for design basis analyses since these show the most limiting initial approach to safety limits, both core thermal and system overpressurization. What is needed to meet the intent of this section of NUREG-0578 is to show the long-term consequences using realistic assumptions (better estimate modeling) incorporating the effects of the following:

1. Operator's failure to act when required.

2. Operator's inappropriate actions during an accident.

3. Additional failures.

4. Selected system operations (e.g., re-starting of RCP's etc.)

The results of these analyses can be used to evaluate information available to the operator and the adequacy of existing procedures. Appropriate changes can then be incorporated into the existing procedures, designs, and training programs. Development of the models to incorporate such effects is in itself a long-term effort before detailed analyses can be run. Significant interaction between industry and the NRC is required to agree on the assumptions, bases, appropriate actions or misactions to be modeled, and best estimate boundary conditions. When completed, the analyses results using the better estimate modeling tools can enhance the current operator training programs by providing additional insight into the course of events the operator will likely encounter during a transient.

The Westinghouse Owners Group (WOG) will submit a detailed description of a program to comply with the requirements of item I.C.1 to the NRC in early 1981. The program will identify WOG's previous submittals to the NRC and the additional effort required to obtain full compliance with this item. After approval of the program by the NRC, CPNPP will submit to the NRC a schedule and program, based on the WOG program, for meeting the requirements of item I.C.1.

NUREG-0737 states that reviews of emergency procedures being conducted under the pilot program, item I.C.8. (NUREG-0660), will be discontinued when the long-term program item I.C.9 (NUREG-0660) is made available. Because it is anticipated that the requirements and guidelines of the long-term program will soon be available, CPNPP will comply with the long-term program rather than submitting procedures for review under the pilot monitoring program.

I.C.2 SHIFT AND RELIEF TURNOVER PROCEDURES

Action Plan Requirements:

"Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the on-going and off-going individuals responsible for command of operations in the control room. Supplementary checklists and shift logs should be developed for the entire operations organization, including instrument technicians, auxiliary operators, and maintenance personnel."

- NUREG 0578, Pg. 13

CPNPP Response

A procedure has been established for shift and relief turnover to provide a checklist for reactor operators and shift supervisors. The checklist for auxiliary operators, a system to evaluate the shift turnover, and the implementation of these procedures will be completed prior to fuel load.

I.C.3 SHIFT SUPERVISOR'S RESPONSIBILITIES

Action Plan Requirements:

"Review plant administrative and management procedures. Revise as necessary to assure that reactor operations command and control responsibilities and authority are properly defined. Corporate management shall revise and promptly issue an operations policy directive that emphasizes the duties, responsibilities, and authority and lines of command of the control room operators, the shift technical advisor, and the person responsible for reactor operations command in the control room (i.e., the senior reactor operator)."

- NUREG 0578, Pg. 12

CPNPP Response

Prior to OL and annually thereafter, the Senior Vice President & Principal Nuclear Officer will issue a management directive that emphasizes the primary management responsibility of the Shift Manager for safe operation of the plant under all conditions and that clearly defines his command duties.

Procedures have been established which define the responsibilities and authorities of the Shift Manager and establish lines of succession. These procedures explicitly address the NUREG concerns dealing with retaining breadth of perspective of operational conditions affecting safety and remaining in the control room under accident conditions to direct operational activities. These new provisions will be incorporated into the training program for the Shift Manager.

I.C.4 CONTROL ROOM ACCESS

Action Plan Requirements:

"Review plant emergency procedures, and revise as necessary, to assure that access to the control room under normal and accident conditions is limited to those persons necessary to the safe command and control of operations."

- NUREG 0578, Pg. 13

CPNPP Response

Administrative procedures have been written to formalize policies which allow the Shift Manager, or his designated alternate, to restrict access to the Control Room during both normal operations and emergencies. The procedures establish a clear line of authority, responsibility and succession in the Control Room in the event of an emergency.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

Action Plan Requirements:

"In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels."

- NUREG 0737

CPNPP Response

Procedures have been established to ensure that reports of industry operating experiences distributed either as INPO SEE-IN reports, as vendor technical correspondence, or as NRC Correspondence are assessed, and that applicable reports are distributed directly to appropriate station personnel, including, when appropriate, the Shift Operations Manager and the Nuclear Training Manager. For INPO SEE-In reports and NRC Information Notices, these procedures direct recipients to develop appropriate action plans in response to the recommendations developed from assessed reports and to inform the Performance Improvement Department of their plans. The Performance Improvement Department will monitor recipients' efforts and

related developments until the problems identified and/or the concerns raised by the report processing are adequately addressed. Vendor technical correspondence is screened for operating experience by the Vendor Equipment Technical Information Program and distributed to the cognizant organization for action/information. The cognizant organization must acknowledge review of the vendor technical correspondence. NRC Bulletins and other pertinent NRC assessments of operating experience are screened by Regulatory Affairs for appropriate distribution and forwarded to the cognizant organization for action/information.

Correspondence which requires a written response or a specific action are tracked to completion by Regulatory Affairs.

Procedures have been established to provide a mechanisim for: reporting problems which occur at CPNPP, correcting the condition, evaluating the reports to identify trends, and determining what changes, if any, are required to prevent recurrence. Procedures assign responsibility for ensuring all required corrective actions are completed. Information on these problems are available to management and plant staff through various methods for dissemination of relevant information to their personnel.

Procedures have been developed in both the Operations department and the Training department to ensure that appropriate information is communicated to operations personnel and incorporated into training programs. These procedures include mechanisms for timely dissemination of information when circumstances warrant. The Shift Operations Manager has the responsibility of screening the available information to ensure both that operations personnel are not burdened with extraneous and unimportant information, and that they do not receive conflicting or contradictory information.

A monthly report will be prepared by Performance Improvement Department and distributed to appropriate site personnel for distribution to licensed and other interested personnel. This report will summarize operating experience related information gathered or compiled during each preceding calendar month, which include summaries of: Licensee Event Reports (LER) generated onsite, significant plant events that are not LER events, and completed assessments of industry operating experience reports. The Plant Support report will include brief narrative histories of the status of both units. These two reports will ensure that all interested personnel remain cognizant of recent industry and station operating experience.

I.C.6 PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Action Plan Requirements:

“It is required (from NUREG-0660) that licensees’ procedures be reviewed and revised, as necessary, to assure that an effective system of verifying and correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such

verification in all instances. The procedures adopted by the licensees may consist of two phases – one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3."

- NUREG 0737

CPNPP Response

Procedures will be established which designate the responsibilities and authorities of personnel associated with the control of equipment. The procedures will address the second, independent verification of proper tagging and the return-to-service of equipment by qualified operators. These procedures will be implemented prior to fuel load.

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

Action Plan Requirements:

"Obtain nuclear steam supply system (NSSS) vendor review of low-power testing procedures to further verify their adequacy."

- NUREG 0694, Pg. 14

"Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy."

- NUREG 0694, Pg. 20

CPNPP Response

CPNPP shall have Westinghouse review the low-power testing procedures prior to fuel loading and the power-ascension and emergency procedures prior to full-power operation.

I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL APPLICANTS

Action Plan Requirements:

"Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steam-line break, or steam-generator tube rupture)."

- NUREG 0694, Pg. 21

Also see NUREG-0737.

CPNPP Response

NUREG-0737 states that reviews of emergency procedures being conducted under the pilot monitoring program, item I.C.8. (NUREG-0660), will be discontinued when the long-term

program item I.C.9 (NUREG-0660) is made available. Because it is anticipated that the requirements and guidelines of the long-term program will soon be available, CPNPP will comply with the long-term program rather than submitting procedures for review under the pilot monitoring program.

I.D CONTROL ROOM DESIGN

OBJECTIVE:

"Improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them."

- NUREG 0660, Pg. I.D-1

I.D.1 CONTROL ROOM DESIGN REVIEW

Action Plan Requirements:

"In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants."

- NUREG 0737

CPNPP Response

The Detailed Control Room Design Review (DCRDR) has been completed in accordance with NUREG-0700 and the reports submitted to the NRC. Onsite audits/reviews have been conducted by the NRC. Implementation of the required design changes has been completed. A final environmental survey was conducted for Unit 1 when construction and modifications were completed. Based on the results of the environmental surveys, Unit 2 control boards were compared to Unit 1 control boards to assess any design differences. Supplements to the DCRDR were submitted prior to fuel loads for Unit 1 and Unit 2 respectively.

Procedures have been developed to formalize the ongoing Human Factors Engineering (HFE) program at CPNPP. The major elements of the HFE program include:

a) HFE review of control room design changes which includes the following:

    1. HFE design review of existing "legacy" control room equipment/systems per guidelines contained in NUREG-0700, Revision 0 [1981].

    2. HFE design review of new and replacement plant computer and plant digital control systems per guidelines contained in NUREG-0700, Revision 2 [2002].

b) Multi-discipline team for the assessment, disposition and verification of Training, Operations and Engineering HFE concerns

c) Site specific HFE design guide

d) Function and task analysis of Emergency Response Guidelines (ERGs) and the comparison with control room capabilities

e) Computer Aided Design (CAD) detailed control board layout drawings

f) Standard abbreviations and acronyms for control room labels and procedures

The DCRDR and the ongoing HFE program were audited by the NRC in January, 1989.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Action Plan Requirements:

"In accordance with Task Action Plan 1.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and license shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status."

- NUREG 0737

Also see NRC Letter dated August 1, 1980.

CPNPP Response

A plant safety parameter display console has provided for CPNPP to assess plant safety status. This system is designed in accordance with NUREG-0696 and NUREG-0737 Supplement 1.

I.G PREOPERATIONAL AND LOW POWER TESTING

OBJECTIVE:

“Increase the capability of the shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events if conducted. Near-term operating license facilities will be required to develop and implement intensified training exercises during the low-power testing programs. This may involve the repetition of startup test on different shifts for training purposes. Based on experiences form the near-term operating license facilities, requirements may be applied to other new facilities or incorporated into the plant drill requirement (Item I.A.2.5). Review comprehensiveness of test programs.”

- NUREG 0660, Pg. I.G-1

I.G.1 TRAINING REQUIREMENTS

Action Plan Requirements:

“Define and commit to a special low-power testing program approved by NRG to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.”

- NUREG 0694, Pg. 15

“Supplement operator training by completing the special low-power test program. Test may be observed by other shifts or repeated on other shifts to provide training to the operators.”

- NUREG 0694, Pg. 21

CPNPP Response

Refer to revised section 13.2.1.1. The referenced program is consistent with 10 CFR 55 rulemaking issued after the TMI-2 event.

Testing to demonstrate the length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance have been performed at the Sequoyah 1 and Salem 2 facilities, which are comparable prototype plants, and will not be performed at CPNPP.

A description of the natural circulation demonstrations that are to be conducted as part of the training program during the low-power test program will be included in the CPNPP FSAR (Chapter 14 - Initial Test Program).

II SITING AND DESIGN

II.B CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW

OBJECTIVE:

"Enhance public safety and reduce individual and societal risk by developing and implementing a phased program to include in safety reviews, consideration of core degradation and melting beyond the design basis. The program phases are (1) short- and medium-term actions for scoping and implementation; (2) added requirements for high population density sites; (3) research programs and design studies to develop additional needed information; and (4) a rulemaking proceeding to establish long-term policy, goals, and requirements related to accidents involving core damage greater than the present design basis."

- NUREG 0660, Pg. II.B-1

II.B.1 REACTOR COOLANT SYSTEM VENTS

Action Plan Requirements:

"Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

"Each licensee shall provide the following information concerning the design and operation of the high point vent system:*

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage."

"*It was the intent of the October 30, 1979 letter to delete the requirement to meet the criteria of 10 CFR 50.44 and SRP 6.2.5 for beyond-design-basis events. The analysis requirements of Position 2 in the September 13, 1979 letter are therefore unnecessary."

CPNPP Response

CPNPP has the capability to remotely vent gases from the reactor vessel head and pressurizer vapor space during a post-accident situation where large quantities of non-condensible gas may collect.

Venting of the reactor vessel head is provided via a line isolated by two one-inch valves in series which vents directly to containment. The venting of the pressurizer vapor space is provided in the same manner. The valves are fail-closed solenoid valves which are environmentally and seismically qualified. The head vent valves are powered by Train A 1E power and the pressurizer vent valves are powered by Train B 1E power.

These vents are operable from the Control Room and positive indication of valve position is also provided in the Control Room.

Procedures will be developed to address the use of the Reactor Coolant System vents. The procedures will address information available to the operator and instructions for initiating or terminating vent usage. This will be implemented upon staff approval.

Details of the design and supporting information required by NUREG- 0737 is provided in Section 1.7, 3.10N, 3.11N, 3A, 5.1 and 17A.

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POSTACCIDENT OPERATION

Action Plan Requirements:

"With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems."

"Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility."

CPNPP Response

A design review of CPNPP radiation and shielding for post-accident operations has been performed in accordance with the guidelines of NUREG-0737. The plant shielding review is a non-nuclear safety-related study, and is not subject to Quality Assurance. The review considered the potential radiation exposure to operators in vital areas and to Class 1E equipment. The following section describes the assumptions and methodology employed in the review and a summary of the results. Any design changes resulting from the study are covered by the

QA requirements for the items being changed. Any additions or modifications to shielding during operations are subject to an appropriate Operations QA Program. The results of the review of radiation qualification of Class 1E equipment will be included in FSAR Section 3.11.

II.B.2.1 Source Terms

The source terms used in this evaluation of shielding are based on the postulated post-accident release of radioactivity equivalent to that described in Regulatory Guide 1.4 and TID-14844.

The reactor core inventory of radioisotopes is derived from the assumptions listed in Table 12.2-24.

The source terms and corresponding systems employed in the evaluation are described in Section 12.2.1.3. The specific subsections are 12.2.1.3.3 for nondepressurized primary coolant, 12.2.1.3.4 for depressurized containment sump water, 12.2.1.3.5 for Process Sampling System sources, and 12.2.1.3.1 for containment atmosphere. These source terms are based on the core activity release fractions identified in Table 12.2-16.

II.B.2.2 System Review

After establishing the source terms noted above, the systems which may contain highly radioactive materials in a post-accident situation have been evaluated. The specific systems reviewed are described in the following subsections.

II.B.2.2.1 Containment

Direct radiation penetrating through the containment walls and radiation streaming through penetrations was based on the source term discussed in Section 12.2.1.3.1. Radiation streaming causes a significant dose rate outside the containment equipment hatch in a field within ″ 25° from the hatch centerline. The dose rate changes with time and distance. Immediately following a LOCA the dose rate is calculated to be about 44 rem per hour at a point 30 feet from the containment surface and about 4 rem per hour at 240 feet. These dose rates decrease to less than 2 rem per hour and 0.06 rem per hour respectively at 24 hours after the accident.

II.B.2.2.2 Emergency Core Cooling System

Plant shielding has been evaluated considering the Emergency Core Cooling System (ECCS) to be operating in the recirculation mode. The residual heat removal pumps take suction from the containment sump, and the safety injection pumps and centrifugal charging pumps take suction from the residual heat removal pump discharge.

Prior to the start of recirculation, the ECCS will contain water from the refueling water storage tank. Although recirculation operation is not initiated until approximately 10 minutes to several hours after the start of an accident the shielding review has assumed that at the beginning of the accident the ECCS contains the source terms described in Section 12.2.1.3.4.

II.B.2.2.3 Residual Heat Removal System

Plant shielding has been evaluated considering the Residual Heat Removal System (RHRS) contains the source terms specified in Section 12.2.1.3.3 for nondepressurized LOCA conditions, or the source terms specified in Section 12.2.1.3.4 for depressurized LOCA conditions as appropriate.

II.B.2.2.4 Containment Spray System

Plant shielding has been evaluated considering the Containment Spray System (CSS) to be operating in the recirculation mode. Prior to the start of recirculation, the CSS will contain water from the refueling water storage tank. Although CSS recirculation operation is not initiated until approximately 15 minutes to several hours after an accident (if required), the shielding evaluation has assumed that at the beginning of the accident the CSS contains the source terms described in Section 12.2.1.3.4.

II.B.2.2.5 Chemical And Volume Control System

A portion of the chemical and volume control system (CVCS), namely the charging pumps and the connecting piping to the RHRS and safety injection system, may contain post-accident sources during depressurized LOCA conditions. Source terms in this portion of the CVCS are described in Section 12.2.1.3.4. The remainder of the CVCS is not expected to become highly radioactive in a post-accident situation because the letdown system is automatically isolated and is not required for accident mitigation.

II.B.2.2.6 Gas Waste Processing System

The Gas Waste Processing System (GWPS) is not expected to contain highly radioactive material generated following an accident since the system is automatically isolated from components that do contain the radioactive material, and it is not required to mitigate an accident. Therefore, the system has been excluded as a potential source of highly radioactive material following a LOCA.

II.B.2.2.7 Post-Accident Sampling System

The Process Sampling System (PSS) is used for post-accident sampling.

Plant shielding has been evaluated considering PSS components and sample lines containing liquid sources as described in Section 12.2.1.3.3 or 12.2.1.3.4.

II.B.2.3 Shielding Methods

Evaluation of plant shielding for equipment qualification dose evaluations following an accident includes direct radiation from the containment, and radiation from piping and components of systems discussed in Section II.B.2.2, as appropriate. Direct radiation through shield walls and streaming and scattered radiation through penetrations were considered. Direct radiation and streaming radiation dose rates for the post-accident shielding evaluation were calculated by point-kernel shielding codes. Exponential attenuation through the shields with dose build-up factors corresponding to the number of mean free paths in the shield were considered. The scattered radiation dose rate from surrounding surfaces was determined by the albedo method.

Gamma energy levels of the radiation source are typically divided into eight or ten energy groups for the analysis.

II.B.2.4 Design Review

The plant shielding design review has identified the radiation exposure rates in vital areas requiring personnel access following a LOCA. The vital areas of the plant which may need periodic access after a LOCA for performing manual operations to mitigate the consequences of the accident are described in Table II.B.2-4. This table also presents the operator path or route numbers, the time after LOCA when access to the area is required, and the calculated radiation dose that may be received by the operator during the performance of the assigned task.

Plant drawings have also been developed to illustrate the post LOCA access routes to the vital areas. These drawings are presented in Figures II.B.2-25 through II.B.2-86. Radiation dose rates between various points of the operator path, the time spent between consecutive points and the path or route number are shown in Table II.B.2-5 and II.B.2-6.

The Control Room Complex and the Technical Support Center (TSC) are vital areas requiring full-time occupancy during the course of an accident. The integrated dose for these areas is less than 5 rem whole-body, or equivalent, for the duration of the accident in accordance with GDC 19. Further details pertaining to the Control Room and TSC dose analyses are provided in Section 15.6.5.4.4.

Motor control centers (MCCs) are not considered to be vital areas for purpose of this review, since single failure criteria were assumed; and a single failure would not prohibit the ability of the MCCs' functions to be performed from the control room.

Access to radwaste control panels is not considered to be vital, since the safety injection signal during postulated DBA conditions will isolate reactor coolant letdown. Therefore, no highly radioactive post-accident fluids will be present in the radwaste systems.

II.B.2.5 Radiation Qualification of Class 1E Equipment

The post-accident radiation exposure to Class 1E equipment located outside and inside containment has been evaluated for post-accident sources postulated to exist following a LOCA. The source terms for equipment located outside containment are described in Section 12.2.1.3.3, 12.2.1.3.4, or 12.2.1.3.5, as applicable. For Class 1E equipment inside containment, the larger dose resulting from sources described in either Section 12.2.1.3.3 or Sections 12.2.1.3.1 and 12.2.1.3.4 were utilized. Equipment dose evaluations for Class 1E equipment inside and outside containment are based on the guidelines of NUREG 0588 and Regulatory Guide 1.89. These dose evaluations consider both gamma and beta radiation, as appropriate for post-accident conditions, and include contributions from normal operations. The calculated post-accident radiation dose values for Class 1E equipment are included in FSAR Appendix 3A.

II.B.3 POST-ACCIDENT SAMPLING

Action Plan Requirements:

"A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less

than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet criteria.

“A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

“In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).”

CPNPP Response

II.B.3.1 General Description

FSAR Section 9.3.2 describes the process sampling system installed in CPNPP. This system was designed for the capability of obtaining samples during normal operating conditions. During postulated accident conditions, radiation levels would be significantly increased.

The Post-Accident Sampling System (PASS) is comprised of two independent subsystems; the Reactor Coolant PASS (RC PASS) and the Containment Air PASS (CA PASS). Figures II.B.3-1 and Flow Diagram M1(2)-301-A illustrate the basic system flow diagrams for the RC PASS and CA PASS respectively.

II.B.3.2 Design Criteria and Functional Requirements

The requirements of NUREG-0737, II.B.3 have been eliminated for CPNPP in accordance with License Amendment 91.

II.B.3.3 PASS Equipment

As stated above, the CPNPP PASS is comprised of two independent subsystems, the Reactor Coolant PASS (RC PASS) and the Containment Air (CA PASS). A discussion of the major components of each of these subsystems is provided below.

II.B.3.3.1 Reactor Coolant PASS

The RC PASS, as shown in Figure II.B.3-1, includes the following major components:

- RC PASS Sample Module
- RC PASS Remote Operating Module
- RC PASS Flush and Diversion Manifold
- RC PASS Auxiliary Module
- RC PASS Flush Module
- RC PASS Sample Cooler

The RC PASS is in layup.

II.B.3.3.2 Containment Air PASS

The Containment Air PASS (CA PASS) will be used to obtain micro-volume samples of the containment atmosphere. The CA PASS, as shown in Flow Diagram

M1(2)-301-A, includes the following major components:

- CA PASS Sample Module
- CA PASS Remote Operating Module
- CA PASS Sample Diversion Valve

The CA PASS is in layup.

II.B.3.3.3 PASS Isolation Valve Control

II.B.3.3.3.1 PASS Isolation Valve Control Panel

The PASS Isolation Valve Control Panel (PIVCP) is a Class 1E, Seismic Category I electrical control panel. It is located in the cable spreading room, a low radiation area.

The PIVCP is provided to permit control for all isolation valves associated with both the RC PASS and the CA PASS. The bypass function of the PIVCP has been disabled.

The PIVCP has control switches for the operation of twelve (12) isolation valves. Three (3) of these valves can also be controlled during normal operation of the Sample Valve Control Panel (SVCP).

Train oriented selector switches are provided at the PIVCP to select controls at the SVCP or PIVCP. The remaining nine (9) valves are controlled exclusively at the PIVCP.

The PIVCP is under direct administrative control of the control room operator. Train oriented control switches configured “CLOSE-LOCAL” are provided in the control room.

Placing these switches in the “LOCAL” position permits operation of the 12 isolation valves from the PIVCP.

II.B.3.3.3.2 Instrument Air to Containment Isolation Valve Control

Instrument air must be available for operation of the PSS Isolation Valves. Instrument air to Containment valve HV-3487 is isolated by a Containment Isolation Phase A signal. This valve must be opened to admit instrument air to tubing run inside containment in order to obtain a sample from RCS.

Opening valve HV-3487 is accomplished by operating the control switch for this valve on Main Control panel CB-01 to the OPEN position. This is a spring-return position in which the switch must be manually held to over-ride the Containment Isolation signal. Releasing the switch will cause the valve to close if the Containment Isolation signal is still present. This operation is in accordance with FSAR Section 7.3.1.1.4.2 Containment Isolation System.

Operating pressure for Instrument air is greater than the maximum pressure in Containment following an accident. The volume of Containment is so great that the capacity and use of the Instrument Air system will not significantly increase Containment pressure. A check valve inside Containment will close if the Containment pressure is greater than the Instrument Air system pressure. Since it meets containment isolation requirements and does not bring reactor coolant or containment atmosphere out of containment post-accident, instrument air is not an isolated auxiliary system (e.g. CVCS letdown) and may be used for this function.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

<u>Action Plan Requirements:</u>

“A program is to be developed to ensure that all operating personnel are training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program should include the following topics.”

"1. Incore Instrumentation

a. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

b. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

c. Methods for calling up (printing) incore data from the plant computer.”

"2. Excore Nuclear Instrumentation (NIS)

a. Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes.”

"3. Vital Instrumentation

a. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs indicated level).

b. Alternative methods for measuring flows, pressures, levels, and temperatures.

1) Determination of pressurizer level if all level transmitters fail.

2) Determination of letdown flow with a clogged filter (low flow).

3) Determination of other Reactor Coolant System parameters if the primary method of measurement has failed."

"4. Primary Chemistry

a. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

b. Expected isotopic breakdown for core damage; for clad damage.

c. Corrosion effects of extended immersion in primary water; time to failure."

"5. Radiation Monitoring

a. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

b. Methods of determining dose rate inside containment from measurements taken outside containment."

"6. Gas Generation

a. Methods of H_2 generation during an accident; other sources of gas (Xe, Ke); techniques for venting or disposal of non-condensibles.

b. H_2 flammability and explosive limit; sources of 0_2 in containment or Reactor Coolant System.

- NRC Letter dated March 28, 1980,
ENCLOSURE 3

"Complete the training of all operating personnel in the use of installed systems to monitor and control accidents in which the core may be severely damaged."

- NUREG 0694, Pg. 22

Also see NUREG 0737

CPNPP Response

Refer to revised sections 13.2.1.1 and 13.2.1.2. The referenced program is consistent with 10 CFR 55 rulemaking issued after the TMI-2 event.

TABLE II.B.2-1
THIS TABLE HAS BEEN DELETED

TABLE II.B.2-2
THIS TABLE HAS BEEN DELETED

TABLE II.B.2-3
THIS TABLE HAS BEEN DELETED

TABLE II.B.2-4
VITAL AREA ACCESS

(Sheet 1 of 5)

	Vital Access Area		Description of Task[1]	Time After LOCA (hrs) When Access is Required	UNIT 1 Route No.	UNIT 1 Operator Dose[7] (rem)	UNIT 2 Route No.	UNIT 2 Operator Dose[7] (rem)
1.	Room No. 78 and adjacent areas, Process Sampling System	A	To lineup the valves in the PSS in the Appropriate Positions so that the required Post Accident samples can be obtained.	0.75	1A Fig. II.B.2-25	2.4	1A Fig. II.B-2.43 Fig. II.B-2-44	2.4
		B	Sample retrieval and transport to the hot lab for analysis.	1.0	1B Fig. II.B.2-26	4.8 (WB) 7.4 (EX)	1B Fig. II.B.2-45 Fig. II.B.2-46	4.7 (WB) 7.4 (EX)
	Room No. 30 and Hot lab	C	To ready the hot lab, the count room and process the samples	0.5	1C Fig. II.B.2-27	0.53 (WB) 1.8 (EX)	1C Fig. II.B.2-47	0.13 (WB) 1.4 (EX)
2.	Room No. 107 Wide Range gas monitoring (WRGM) sample skid		To retrieve sample from the plant Post LOCA effluent being released through the stacks, using the WRGM system X-RE-5570 A&B	Various times are needed starting at t=30 min.	2 Fig. II.B.2-30	0.21[2] (WB) 0.4 (EX)	2 Fig. II.B.2-50 Fig. II.B.2-51	0.21[2](WB) 0.4 (EX)
3.	Room No. 245 Mechanical equipment room CCW surge tank area		To manually adjust the supply valves to fill the CCW surge tank, in case of an instrument air malfunction[3]	Various times as needed starting at t=30	3 Fig. II.B.2-31	2.8[2]	3 Fig. II.B.2-52 Fig. II.B.2-53	2.8[2]

TABLE II.B.2-4
VITAL AREA ACCESS

(Sheet 2 of 5)

	Vital Access Area	Description of Task[1]	Time After LOCA (hrs) When Access is Required	UNIT 1 Route No.	UNIT 1 Operator Dose[7] (rem)	UNIT 2 Route No.	UNIT 2 Operator Dose[7] (rem)
4.	Room No. 245 Mechanical equipment room, Safety chilled water valves	To manually throttle valves to fill the safety chilled water surge tank in case of an instrument air malfunction[3]	Various times as needed starting at t=30	4 Fig. II.B.2-31	2.8[2]	4 Fig. II.B.2-52 Fig. II.B.2-53	2.8[2]
5.	Room No. 115A, or 115B Safety chiller equipment area	To adjust the pressure controller valve on the safety chiller condenser refrigerant in case of an instrument air malfunction.[3]	0.5	5 Fig. II.B.2-32	0.05	5 Fig. II.B.2-54 Fig. II.B.2-55	0.07
6.	Room No. 72, 73, and 74, AFW Pump rooms	To manually throttle the auxiliary feedwater flow or cycle the AFW pumps to control S/G level.	0.5	6 Fig. II.B.2-33	2.2[3]	6 Fig. II.B.2-56 Fig. II.B.2-57	2.2[3]
7.	Room No. 174[4] Laundry holdup area, Reactor makeup water pump area	To switch over the reactor makeup water pumps in case of a low pressure alarm	Between 2 min. to 30 min. after LOCA	7 Fig. II.B.2-34	0.04	7 Fig. II.B.2-58 Fig. II.B.2-59	0.43
8.	Room No. 174[4] Laundry holdup area, Reactor makeup water pump area	To isolate the non-nuclear safety-related portion of the reactor makeup water system	Between 2 to 30 min. after LOCA Fig. II.B.2-59	8 Fig. II.B.2-34	0.04	8 Fig. II.B.2-58	0.43

TABLE II.B.2-4
VITAL AREA ACCESS

(Sheet 3 of 5)

				UNIT 1		UNIT 2	
	Vital Access Area	Description of Task[1]	Time After LOCA (hrs) When Access is Required	Route No.	Operator Dose[7] (rem)	Route No.	Operator Dose[7] (rem)
9.	Room No. 247A, Fuel pool cooling pipe tunnel	To manually maintain water level in the spent fuel pool	17.0	9 Fig. II.B.2-35	0.01	9 Fig. II.B.2-60 Fig. II.B.2-61	0.01
10.	Rooms No. 84 and 85[5] Diesel generator control station	To inspect the fuel oil storage tank level, fill the oil storage tanks when needed	1.0	10 Fig. II.B.2-36	0.86	10 Fig. II.B.2-62 Fig. II.B.2-63 Fig. II.B.2-64	0.84
11.	Rooms No. 84 and 85[5] Diesel generator control station	To inspect the DG lube oil makeup duplex filter and strainers	1.0	11 Fig. II.B.2-36	0.86	11 Fig. II.B.2-62 Fig. II.B.2-63 Fig. II.B.2-64	0.84
12.	Equipment Yard Diesel generator fuel oil tank filling area	To fill main diesel generator oil storage tanks from a delivery tanker truck	Three days after LOCA[8]	12 Fig. II.B.2-37	2.2[9]	12 Fig. II.B.2-65	2.2[9]
13.	Task deleted						
14.	Rooms No. 82, 95, 96, 103, 113, 133, 134 and 241	To deenergize lights, in the case of loss of the non ESF ventilation after a LOCA to prevent temperature excursions	0.5	14 Fig. II.B.2-39	0.61	14 Fig. II.B.2-68 through Fig. II.B.2-73	0.64

TABLE II.B.2-4
VITAL AREA ACCESS

(Sheet 4 of 5)

				UNIT 1		UNIT 2	
	Vital Access Area	Description of Task[1]	Time After LOCA (hrs) When Access is Required	Route No.	Operator Dose[7] (rem)	Route No.	Operator Dose[7] (rem)
15.	Task deleted.						
16.	Room No.79 Safeguards Drain Panel (TAG No. CPI-E1PRLV-24)	Diagnose passive failures in the ESF Systems.	24.0	16 Fig. II.B.2-41	0.11	16 Fig. II.B.2-78	0.07
17.	Rooms No.150 and 150A, Mechanical equipment area	To manually open control room intake dampers.	Within 48	17	0.3[6] Fig. II.B.2-42	17 Fig. II.B.2-79 Fig. II.B.2-80	0.30[6]
18.	Control Room and Technical Support Center	General responsibilities of the Control Room and the TSC personnel	All times	N/A	Refer to Section 15.6.5.4.4.	N/A	Refer to Section 15.6.5.4.4
19.	Rooms 213, 249 249A, 249B, 264 and 268	Manually realign valves to maintain spent fuel pool cooling	0.5	19 Fig. II.B.2-42.1 Fig. II.B.2-42.2 Fig. II.B.2-42.3	1.5	19 Fig. II.B.2-81 Fig. II.B.2-82 Fig. II.B.2-83 Fig. II.B.2-84	1.9

TABLE II.B.2-4
VITAL AREA ACCESS

(Sheet 5 of 5)

				UNIT 1		UNIT 2	
	Vital Access Area	Description of Task[1]	Time After LOCA (hrs) When Access is Required	Route No.	Operator Dose[7] (rem)	Route No.	Operator Dose[7] (rem)
20.	Rooms 108A and 108B	Manually isolate steam supply to the AFWPT	7.0	20 Fig. II.B.2-42.4	4.0[10]	20 Fig. II.B.2-85 Fig. II.B.2-86	4.1[10]

Notes:

(1) It is assumed that each task is performed by one operator (except for Task 20).

(2) This is the maximum dose which occurs due to buildup on ESF filters at 168 hours after LOCA.

(3) These tasks may have to be repeated several times and more than one operator may be needed.

(4) Tasks No. 7 and 8 may be performed by the same operator.

(5) Task Nos. 10 and 11 may be performed by the same operator.

(6) This is the maximum dose which occurs due to buildup on Control Room pressurization filters and intake air inducting upstream of the filters at 8 hours after LOCA.

(7) Values listed represent total accumulated radiation dose to the whole body and whole body dose = extremity dose, unless separate values are noted by WB (Whole Body) or EX (Extremity). Values include all applicable dose components (e.g., task standby/briefing, en route transit, task performance, handling of radioactive materials.)

(8) Although tank is not required to be filled for one week, operators may choose to begin filling the tank after 3 days.

(9) This is the maximum dose which assumes access plate removal and replacement for each tanker truck delivery. In addition, one operator is assumed to perform the task for all deliveries.

(10) Two operators are required to perform this Task. Access is required to each room, 108A and 108B, (one separate operator per room). Dose reflects operator dose to either room.

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 1 of 12)

A = letters correspond to points on operator access route
t = time (seconds) between consecutive points
dr = dose rate (mrem/hr) between consecutive points
task = vital operator action

		A		B		C		D		E		F		G	
Operator Task and Route No.		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post accident sampling														
	Route #1A Initial Valve Lineup	13	27	26	27	11	190	38	2,700	7	39,000	15	16,000	4	36,100
	Route #1B sample retrieval	13	27	26	27	11	190	38	2,700	7	39,000	15	16,000	30	36,100
	Route #1C preparation of hot lab	13	34	26	34	-	1.1	-	34	- Task 120 min.	230				
2.	Sampling of plant gaseous release from stack WRGM Route #2	13	0.3	26	0.3	11	26	3	23	14	2.5	13	5.2	37	2.5
3.	Filling the CCW surge tank Route #3	13	0.3	33	0.3	14	2.5	13	5.2	37	2.5	29	5.0	64	49
4.	Filling the safety chilled Water surge tank Route #4	13	0.3	33	0.3	14	2.5	13	5.2	37	2.5	29	5.0	64	49
5.	Safety chilled water chiller valve adjustment Route #5	13	77	33	34	14	4.1	13	305	37	4.1	15	5.0	23	28
6.	Adjusting the AFW flow Route #6	13	77	33	34	14	4.1	13	305	37	4.1	15	5.0	23	28
7.	Switch over the reactor makeup water pump Route #7	13	77	33	34	14	4.1	13	305	37	4.1	15	5.0	23	28

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 2 of 12)

Operator Task and Route No.	A		B		C		D		E		F		G	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
8. Isolation of NNS portion of the reactor makeup water system Route #8	13	77	33	34	14	4.1	13	305	37	4.1	15	5.0	23	28
9. Maintaining the fuel pool water level Route #9	13	18	33	18	14	3.3	13	170	37	3.3	15	5.0	23	17
10. Inspection of the diesel generator fuel tanks level Route #10	13	27	33	27	14	3.7	13	253	37	3.7	15	5.0	6	5.0
11. Inspection of the diesel generator lube oil makeup duplex filters and strainers Route #11	13	27	33	27	14	3.7	13	253	37	3.7	15	5.0	6.0	5.0
12. Filling the main diesel generator fuel storage tank Route #12	54	0.3	60	0.6	90	0.5	30	125	- Task 3 hrs[2]	125				
13. Task deleted														
14. Deenergizing the lights Route #14	46	77	14	4.1	67 Task 60 sec.	3,300	14	4.1	13	305	37	4.1	29	5.0
15. Task deleted.														
16. Safeguards Drain Panel Access Route #16	13	1.0	33	1.0	11	1.0	24	108	7	2,000	-	2,000 Task 3 min.	-	-

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 3 of 12)

	A		B		C		D		E		F		G	
Operator Task and Route No.	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
17. Manually open Control Room HVAC Intake Damper Route #17	30	1.0	3	24.8	20	1.66	10	24.8	15 Task 30 min.	565	-	-	-	-
19. Manually realign valves to maintain spent fuel pool cooling	12.5	77	33	34	14.3	4.1	13.3	305	36.7	4.1	16.7	5	47	5
20. Manually isolate steam supply to the AFWPT	12.5	18	33	18	14.3	3.3	13.3	170	36.7	3.3	29.3	5	64.3	24.7

	H		I		J		K		L		M	
Operator Task and Route No.	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1. Route 1A (contd)	6	36,100	- Task 3.7 min.	18,000	6	18,000	10	27,000	20	22,300	20	22,300
Route 1B (contd)	- Task 6 min.	16,200	- Task 2 min.	36,100								
Route 1C (contd)												
2. Sampling of plant (contd)	29	5.0	64	49	15	7.0	10	4.4	15 Task 18 min.	600		
3. Filling the CCW (contd)	15	7.0	12	4.4	6	300	Task 65 min.	2,600				
4. Filling the safety (contd)	15	7.0	12	4.4	6	300	Task 65 min.	2,600				

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 4 of 12)

		H		I		J		K		L		M	
Operator Task and Route No.		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
5.	Safety chilled water (contd)	17	66	9	125	9	740	26 Task 60 min.	5.0				
6.	Adjusting the AFW (contd)	17	66	9	125	9	740	5	2,600	6	20,000	20	81,000
7.	Switch over (contd)	30 Task 20 min.	4.1										
8.	Isolation of NNS (contd)	30 Task 10 min.	4.1										
9.	Maintaining the fuel (contd)	30	3.3	28	380	7 Task 97 min.	2.5						
10.	Inspection of diesel (contd)	38	3,600	10	280	19	280	22	548	7	890	15	38,000
11.	Inspection of diesel (contd)	38	3,600	10	280	19	280	22	548	7	890	15	38,000
12.	Filling the main diesel (contd)												
13.	Task deleted												
14.	Deenergizing the lights (contd)	97 Task 90 sec.	24.7	15	5.0	64	5.0	31	4,400	13 Task 30 sec.	6,500	8	4,400
15.	Task deleted.												
16.	Safeguards Drain Panel Access (contd)												

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 5 of 12)

Operator Task and Route No.	H		I		J		K		L		M	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
17. Manually open control room HVAC Intake Damper (contd)												
19. Manually realign valves to maintain spent fuel pool cooling	39.6	3510	126 Task 10 min.	5580	39.6	3510	47	5	31.3	5	23.3	28
20. Manually isolate steam supply to the AFWPT	14.7	89.5	12	42	9.3	250	13	96000	- Task 124 sec.			

Operator Task and Route No.	N		O		P		Q		R		S		T	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1. Post accident (contd)														
2. Sampling of plant (contd)														
3. Filling the CCW (contd)														
4. Filling the safety (contd)														
5. Safety chilled water (contd)														
6. Adjusting the AFW (contd)	8	29,000	6	1,240	4	13,300	9	13,000	3 Task 15 min.	1,900				
7. Switch over (contd)														
8. Isolation of NNS (contd)														
9. Maintaining the fuel (contd)														

TABLE II.B.2-5
DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 6 of 12)

Operator Task and Route No.	N		O		P		Q		R		S		T	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
10. Inspection of diesel (contd)	12	38,000	11	6,200	16 Task 5 min.	5.0	19	6,200	18	9,800	16 Task 5 min.	5.0		
11. Inspection of diesel (cont)	12	38,000	11	6,200	16 Task 10 min.	5.0	19	6,200	18	9,800	16 Task 10 min.	5.0		
12. Filling the main diesel (contd)														
13. Task deleted														
14. Deenergizing the lights (contd)	22 Task 30 sec.	350	24.3	686	7.3	1,100	15	8,500	10 Task 30 sec.	17,700	12	10,800	27 Task 15 sec.	410
15. Task deleted.														
16. Safeguards Drain Panel Access (contd)														
17. Manually open Control Room HVAC Intake Damper (contd)														
19. Manually realign valves to maintain spent fuel pool cooling	30.3	4.1	27.8	380	25	2.5	13.3	25	77.3 Task 32 min.	25	96.6 Task 28 min.	398	42 Task 5 min.	25
20. Manually isolate steam supply to the AFWPT														

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 7 of 12)

Operator Task and Route No.	U		V		W		X		Y		Z	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1. Post accident (contd)												
2. Sampling of plant (contd)												
3. Filling the CCW (contd)												
4. Filling the safety (contd)												
5. Safety chilled water (contd)												
6. Adjusting the AFW (contd)												
7. Switch over (contd)												
8. Isolation of NNS (contd)												
9. Maintaining the fuel (contd)												
10. Inspection of diesel (contd)												
11. Inspection of diesel (contd)												
12. Filling the main diesel (contd)												
13. Task deleted												
14. Deenergizing the lights (contd)	19 Task 30 sec.	1,120	22	410	12	10,800	10	17,700	15	8,500	6	1,100
15. Task deleted.												
16. Safeguards Drain Panel Access (contd)												

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 8 of 12)

	U		V		W		X		Y		Z	
Operator Task and Route No.	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
17. Manually Open Control Room HVAC Intake Damper (contd)												
19. Manually realign valves to maintain spent fuel pool cooling	13.3	25	25	2.5	27.8	380	30.3	4.1	23.3	28	14.7	5
20. Manually isolate steam supply to the AFWPT												

	AA		BB		CC		DD		EE		FF		GG	
Operator Task and Route No.	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1. Post accident (contd)														
2. Sampling of plant (contd)														
3. Filling the CCW (contd)														
4. Filling the safety (contd)														
5. Safety chilled water (contd)														
6. Adjusting the AFW (contd)														
7. Switch over (contd)														
8. Isolation of NNS (contd)														
9. Maintaining the fuel (contd)														
10. Inspection of diesel (contd)														

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 9 of 12)

Operator Task and Route No.	AA t	AA dr	BB t	BB dr	CC t	CC dr	DD t	DD dr	EE t	EE dr	FF t	FF dr	GG t	GG dr
11. Inspection of diesel (contd)														
12. Filling the main diesel (contd)														
13. Task deleted														
14. Deenergizing the lights (contd)	24.3	686	24.3	350	38	4,400	64	5.0	15	5.0	37	4.1	13	305
15. Task deleted.														
16. Safeguards Drain Panel Access (contd)														
17. Manually Open Control Room HVAC Intake Damper (contd)														
19. Manually realign valves to maintain spent fuel pool cooling	36.7	4.1	13.3	305	14.3	4.1	33	34	12.5	77				
20. Manually isolate steam supply to the AFWPT														

Operator Task and Route No.	HH t	HH dr	II t	II dr	JJ t	JJ dr	KK t	KK dr	LL t	LL dr	MM t	MM dr
1. Post accident (contd)												
2. Sampling of plant (contd)												
3. Filling the CCW (contd)												

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 10 of 12)

Operator Task and Route No.	HH		II		JJ		KK		LL		MM	
	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
4. Filling the safety (contd)												
5. Safety chilled water (contd)												
6. Adjusting the AFW (contd)												
7. Switch over (contd)												
8. Isolation of NNS (contd)												
9. Maintaining the fuel (contd)												
10. Inspection of diesel (contd)												
11. Inspection of diesel (contd)												
12. Filling the main diesel (contd)												
13. Task deleted												
14. Deenergizing the lights (contd)	14	4.1	26	34	56 Task 30 sec.	5.0	6	1.7	45 Task 30 sec.	5.0	6	1.7
15. Task deleted.												
16. Safeguards Dring Panel Access (contd)												
17. Manually Open Control Room HVAC Intake Damper (contd)												

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 11 of 12)

Operator Task and Route No.	HH t	HH dr	II t	II dr	JJ t	JJ dr	KK t	KK dr	LL t	LL dr	MM t	MM dr
19. Manually realign valves to maintain spent fuel pool cooling (contd)												
20. Manually isolate steam supply to the AFWPT												

Operator Task and Route No.	NN t	NN dr	OO t	OO dr	PP t	PP dr	QQ t	QQ dr	RR t	RR dr	SS t	SS dr	TT t	TT d
1. Post accident (contd)														
2. Sampling of plant (contd)														
3. Filling the CCW (contd)														
4. Filling the safety (contd)														
5. Safety chilled water (contd)														
6. Adjusting the AFW (contd)														
7. Switch over (contd)														
8. Isolation of NNS (contd)														
9. Maintaining the fuel (contd)														
10. Inspection of diesel (contd)														
11. Inspection of diesel (cont)														
12. Filling the main diesel (contd)														

TABLE II.B.2-5

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 1 OPERATOR ROUTE[1]

(Sheet 12 of 12)

	NN		OO		PP		QQ		RR		SS		TT	
Operator Task and Route No.	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	d
13. Task deleted														
14. Deenergizing the lights (contd)	33	5.0	20	77										
15. Task deleted.														
16. Safeguards Drain Panel Access (contd)														
17. Manually Open Control Room HVAC Intake Damper (contd)														
19. Manually realign valves to maintain spent fuel pool cooling (contd)														
20. Manually isolate steam supply to the AFWPT														

Notes:

1) Dose rates and times are shown for operator transit and task performance only. The total accumulated radiation doses, which also include task standby/briefing and/or handling of radioactive materials, are listed in Table II.B.2-4.

2) This task is assumed to initiate every 12 hours, three days following a LOCA for each of the nine tanker truck deliveries. The dose rates listed are the maximum for any delivery.

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 1 of 14)

A = Letters correspond to points on operator access route.
t = Time (seconds) between consecutive points.
dr = Dose rate (mrem/hr) between consecutive points.
task =Vital operator action

		A		B		C		D		E		F		G	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post Accident Sampling														
1A.	Inital valve lineup	12.5	77	26.4	34	10.4	39	5	5	36	4.1	13.1	305	14.1	4.1
1B.	Sample Retrieval	12.5	27	26.4	27	10.4	36	10	5	72	3.7	26.2	253	28.2	3.7
1C.	Hot lab preparation	12.5	77	26.4	34	5	1.1	20	34	5	39	- Task 120 min	39		
2.	Sampling of plant gaseous release from stack WRGM	12.5	0.3	26.4	0.3	10.4	25	29.3	5	64.3	49	14.7	6.7	9.6	4.4
3.	Filling the CCW surge tank	12.5	0.3	33	0.3	5	2.5	29.3	5	64.3	49	14.7	7	12	4.4
4.	Filling the safety chilled water surge tank	12.5	0.3	33	0.3	5	2.5	29.3	5	64.3	49	14.7	7	12	4.4
5.	Safety chilled water system valve adjustment	12.5	77	33	34	5	5	14.7	5	23.3	28	17.3	66	9.3	125
6.	Adjusting the AFW flow rate	12.5	77	33	34	5	5	14.7	5	5	2600	9.3	740	9.3	125
7.	Switchover the reactor makeup water pumps in case of low pressure alarm	12.5	77	33	34	5	5	14.7	5	23.3	28	14	66	30.3	1040

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 2 of 14)

		A		B		C		D		E		F		G	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
8.	Isolation of NNS portion of the reactor makeup water system	12.5	77	33	34	5	5	14.7	5	23.3	28	14	66	30.3	1040
9.	Maintaining the fuel pool water level	12.5	6	33	6	5	5	14.7	5	23.3	8	17.3	17	30.3	195
10.	Inspection of the diesel generator fuel tank level	12.5	27	33	27	5	5	14.7	5	64.3	5	38.3	3600	9.8	280
11.	Inspection of the diesel generator lube oil makeup duplex filters & strainers	12.5	27	33	27	5	5	14.7	5	64.3	5	38.3	3600	9.8	280
12.	Filling the main diesel generator fuel storage tank	97	0.3	60	0.6	90 Task 3 hrs.[2]	0.5	37.5	125	-	125				
13.	Task Deleted														
14.	De-energizing lights	12.5	77	33	34	14.3 Task 1 min.	4.1	13.3	305	36.7	4.1	66.6	3300	36.7	4.1
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access	12.5	1	33	1	14.3	2.5	13.3	17	36.7	2.5	16	110	9	2000

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 3 of 14)

		A		B		C		D		E		F		G	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
17.	Manually open Control Room HVAC intake damper	12.5	18	9	18	39.3	5	31.6	1.7	7	24.7	20	1.7	12	24.7
19.	Manually realign valves to maintain spent fuel pool cooling	12.5	77	33	34	5	5	16.7	5	21.7	5	39.6	3510	126 Task 10 min	5580
20.	Manually isolate steam supply to the AFWPT	12.5	18	33	18	5	33	29.3	5	64.3	24.7	14.7	89.5	12	42

		H		I		J		K		L		M		N	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post-accident sampling														
1A.	Initial valve lineup	38	3300	7	47000	15	19000	4	43700	6	43700	- Task 3.7 min.	21700	6	21700
1B.	Sample retrieval	76	2700	14	39000	15	16000	30	36100	60	16200	60 Task 2 min.	16200	- Task 4 min.	36100
1C.	Hot lab preparation														
2.	Sampling of plant gaseous release from stack WRGM	14.7	600	- Task 18 min.	600										
3.	Filling the CCW surge tank	6.4	300	- Task 65 min.	2600										

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 4 of 14)

Route	Operator Task	H		I		J		K		L		M		N	
		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
4.	Filling the safety chilled water surge tank	6.4	300	- Task 65 min.	2600										
5.	Safety chilled water system valve adjustment	9.3	740	26.3	5	- Task 60 min.	5								
6.	Adjustng the AFW flow rate	17.3	66	23.3	28	5.8	20000	20	81000	8	20000	6.3	1240	4.3	13300
7.	Switchover the reactor makeup water pumps in case of low pressure alarm	- Task 20 min.	1040												
8.	Isolation of NNS portion of the reactor makeup water system	- Task 10 min.	1040												
9.	Maintaining the fuel pool water level	27.8	380	6.7	2.5	- Task 97 min.	2.5								
10.	Inspection of the diesel generator fuel tank level	19	280	22	548	7	890	14.7	38000	12	38000	11.2	6200	16.1	5
11.	Inspection of the diesel generator lube oil makeup duplex filters & strainers	19	280	22	548	7	890	14.7	38000	12	38000	11.2	6200	16.1	5
12.	Filling the main diesel generator fuel storage tank														
13.	Task Deleted														

TABLE II.B.2-6
DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 5 of 14)

Route	Operator Task	H t	H dr	I t	I dr	J t	J dr	K t	K dr	L t	L dr	M t	M dr	N t	N dr
14.	De-energizing lights	13.3	305	14.3	4.1	29.3	5	96.6 Task 1.5 min.	24.7	14.7	5	64.3	5	30.7	4400
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access	- Task 3 min.	1200												
17.	Manually open Control Room HVAC intake damper	12	561	- Task 30 min.	561										
19.	Manually realign valves to maintain spent fuel pool cooling	39.6	3510	21.7	5	31.3	5	23.3	28	14	66	30.3	1040	27.8	380
20.	Manually isolate steam supply to the AFWPT	9.3	250	13	97000	- Task 124 sec									

Route	Operator Task	O t	O dr	P t	P dr	Q t	Q dr	R t	R dr	S t	S dr	T t	T dr	U t	U dr
1.	Post-accident Sampling														
1A.	Initial valve lineup	10	21700	20	27000	20	27000								

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 6 of 14)

Route	Operator Task	O		P		Q		R		S		T		U	
		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1B.	Sample retrieval														
1C.	Hot lab preparation														
2.	Sampling of plant gaseous release from stack WRGM														
3.	Filling the CCW surge tank														
4.	Filling the safety chilled water surge tank			2.5 2900	- 1900	6 1900									
5.	Safety chilled water system valve adjustment														
6.	Adjusting the AFW flow rate	6	13000	2.5	11500	2.5	1900	- Task 15 min.	1900	7.7	13300	6 Task 15 min.	1900		
7.	Switchover the reactor makeup water pumps in case of low pressure alarm														
8.	Isolation of NNS portion of the reactor makeup water system														
9.	Maintaing the fuel pool water level														
10.	Inspection of the diesel generator fuel tank level	- Task 15 min.	5	2.8	6200	18.3	9800	16.1	5	- Task 15 min.	5				

TABLE II.B.2-6
DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 7 of 14)

Route	Operator Task	O		P		Q		R		S		T		U	
		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
11.	Inspection of the diesel generator lube oil makeup duplex filters and strainers	- Task 10 min.	5	2.8	6200	18.3	9800	16.1	5	- Task 10 min.	5				
12.	Filling the main diesel generator fuel storage tank														
13.	Task Deleted														
14.	De-energizing lights	12.6 Task 0.5 min.	6500	7.7	4400 Task 0.5 min.	22	350	24.3	686	7.5	1100	14.7	8500	10	17700
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access														
17.	Manually open Control Room HVAC intake damper														
19.	Manually realign valves to maintain spent fuel pool cooling	25	2.5	13.3	25 Task 32 min	77.3	25	96.6 Task 28 min	1150	42 Task 5 min	25	13.3	25	25	2.5
20.	Manually isolate steam supply to the AFWPT														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 8 of 14)

		V		W		X		Y		Z		AA		BB	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post-accident Sampling														
1A.	Initial valve lineup														
1B.	Sample Retrieval														
1C.	Hot lab preparation														
2.	Sampling of plant gaseous release from stack WRGM														
3.	Filling the CCW surge tank														
4.	Filling the safety chilled water surge tank														
5.	Safety chilled water system valve adjustment														
6.	Adjusting the AFW flow rate														
7.	Switchover the reactor makeup water pumps in case of low pressure alarm														
8.	Isolation of NNS portion of the reactor makeup water system														
9.	Maintaining the fuel pool water level														
10.	Inspection of the diesel generator fuel tank level														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 9 of 14)

		V		W		X		Y		Z		AA		BB	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
11.	Inspection of the diesel generator lube oi makeup duplex filters and strainers														
12.	Filling the main diesel generator fuel storage tank														
13.	Task Deleted														
14.	De-energizing lights	12	10800	27.3 Task 0.25 min.	410	18.6 Task 0.5 min.	1120	21.7	410	12	10800	10	17700	14.7	8500
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access														
17.	Manually open Control Room HVAC intake damper														
19.	Manually realsign valves to maintain spent fuel pool cooling	27.8	380	30.3	1040	14	66	23.3	28	14.7	5	5	5	33	34
20.	Manually isolate steam supply to the AFWPT														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 10 of 14)

Route	Operator Task	CC		DD		EE		FF		GG		HH		II	
		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post-accident Sampling														
1A.	Initial valve lineup														
1B.	Sample Retrieval														
1C.	Hot lab preparation														
2.	Sampling of plant gaseous release from stack WRGM														
3.	Filling the CCW surge tank														
4.	Filling the safety chilled water surge tank														
5.	Safety chilled water system valve adjustment														
6.	Adjusting the AFW flow rate														
7.	Switchover the reactor makeup water pumps in case of low pressure alarm														
8.	Isolation of NNS portion of the reactor makeup water system														
9.	Maintaining the fuel pool water level														
10.	Inspection of the diesel generator fuel tank level														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 11 of 14)

		CC		DD		EE		FF		GG		HH		II	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
11.	Inspection of the diesel generator lube oi makeup duplex filters and strainers														
12.	Filling the main diesel generator fuel storage tank														
13.	Task Deleted														
14.	De-energizing lights	8	1100	22	686	28.2	350	38.3	4400	64.3	5	14.7	5	26.4	34
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access														
17.	Manually open Control Room HVAC intake damper														
19.	Manually realign valves to maintain spent fuel pool cooling	12.5	77												
20.	Manually isolate steam supply to the AFWPT														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 12 of 14)

		JJ		KK		LL		MM		NN		OO		PP	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
1.	Post-accident Sampling														
1A.	Initial valve lineup														
1B.	Sample Retrieval														
1C.	Hot lab preparation														
2.	Sampling of plant gaseous release from stack WRGM														
3.	Filling the CCW surge tank														
4.	Filling the safety chilled water surge tank														
5.	Safety chilled water system valve adjustment														
6.	Adjusting the AFW flow rate														
7.	Switchover the reactor makeup water pumps in case of low pressure alarm														
8.	Isolation of NNS portion of the reactor makeup water system														
9.	Maintaining the fuel pool water level														
10.	Inspection of the diesel generator fuel tank level														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 13 of 14)

Route	Operator Task	JJ		KK		LL		MM		NN		OO		PP	
		t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
11.	Inspection of the diesel generator lube oi makeup duplex filters and strainers														
12.	Filling the main diesel generator fuel storage tank														
13.	Task Deleted														
14.	De-energizing lights	56.4 Task 0.5 min.	5	9.7	1.7	45.2 Task 1 min.	5	9.7	1.7	33.2	5	14.9	5		
15A.	Task deleted.														
15B.	Task deleted.														
16.	Safeguards drain panel access														
17.	Manually open Control Room HVAC intake damper														
19.	Manually realign valves to maintain spent fuel pool cooling														

TABLE II.B.2-6

DOSE RATES AND TIME SPENT BETWEEN CONSECUTIVE POINTS OF UNIT 2 OPERATOR ROUTE[1]

(Sheet 14 of 14)

		JJ		KK		LL		MM		NN		OO		PP	
Route	Operator Task	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr	t	dr
20.	Manually isolate steam supply to the AFWPT														

Notes:

1) Dose rates and times are shown for operator transit and task performance only. The total accumulated radiation doses, which also include task standby/briefing and/or handling of radioactive materials, are listed in Table II.B.2-4.

2) This task is assumed to initiate every 12 hours, three days following a LOCA for each of the nine tanker truck deliveries. The dose rates listed are the maximum for any delivery.

TABLE II.B.3-1
DELETED

II.D REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES

OBJECTIVE:

"Demonstrate by testing and analysis that the relief and safety valves, block valves and associated piping in the reactor coolant system are qualified for the full range of operating and accident conditions. Anticipated transients without scram (ATWS) may be considered in later phases of the test program. In addition, design changes or modifications will be made that are necessary to provide positive indication of valve position."

- NUREG 0660, Pg II.D-1

II.D.1 TESTING REQUIREMENTS

Action Plan Requirements:

"Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents."

CPNPP Response

CPNPP participated fully in the EPRI Safety and Relief Valve Test Program to demonstrate acceptable valve operability under expected flow conditions. Table II.D.1-1 key plant parameters pertaining to the pressurizer safety, relief, and PORV block valves. Also see Section 5.4.13 for additional details about the safety and relief valves. Valves identical to those installed at CPNPP were tested in the EPRI test program as documented in Reference 1. Block valves tested are documented in Reference 2. Inlet fluid conditions for the tests adequately envelope those expected at CPNPP as documented in Reference 3 and 8. Justification that the actual valve test conditions adequately represent anticipated plant conditions is provided in Reference 4, Sections 3.7 and 4.7 for relief and safety valves respectively. Test results for the CPNPP-representative safety and block valves are summarized in Reference 5, Sections 3.5 and 4.6 respectively.

A. The CPNPP safety valve inlet piping configuration is shorter than that employed in the tests. However, the test conditions report (Reference 4, Section 4.7), states that only the long inlet piping configuration was tested because, "…the operability of such valves with a shorter inlet is expected to be as good or better than that observed with the test configuration." Safety valve ring settings for CPNPP are comparable to the "reference" ring settings of the following tests: 929, 931a, 932, 1406, 1411, 1415, and 1419. Further discussion of the inlet piping and valve ring settings comparison with the EPRI Tests is provided in Reference 8.

   Expected valve discharge back pressure for CPNPP was modeled by Continuum Dynamics for EPRI and the results are presented in Reference 6, page 3-8. The calculated back pressure of 291 psi is well below that of the referenced 900-series of EPRI tests (650 psi) and only moderately above the 245 psi of the 1400-series of tests.

   The relief valve tests results (NP-2628-LD, Section 4.6), reflect acceptable operation in both the Marshall Steam Station tests and the Wyle Phase III tests.

Based upon the above indicated references to the appropriate reports produced in support of EPRI Project V102, the pressurizer safety and relief valves, as installed at CPNPP, were adequately represented by the previously referenced tests. Westinghouse, in Reference 7, concluded that those same tests adequately demonstrated acceptable performance of the tested safety valves for pressurizer relief at CPNPP.

Reanalysis of the CPNPP Unit 1 pressurizer safety and relief valve piping determined it to be adequate provided a loop seal temperature profile similar to that produced in EPRI Test 917 is maintained. Consequently, the safety valve loop seals will be insulated to maintain the referenced or hotter temperature profile. The calculated piping support loads have been incorporated into a reanalysis of the existing piping supports. Based on the EPRI test temperature profile and Westinghouse recommendation, since the presence of insulation on the upstream piping, which slopes to the valve, would prevent heat transfer from the piping and thus promote flashing of the liquid into steam, no insulation is installed upstream of the PORVs. A water seal is maintained at the PORVs via the sloped piping.

B. Block Valves - A small number of block valves, including the type and model used for CPNPP, were tested at the Marshall Steam Station Test Facility. After modifications to the valve operator, the valve successfully closed under high flow and high differential pressure conditions. Sufficient data was gathered to determine the modifications required to assure correct valve operation under all fluid conditions.

Subsequently, in response to IE Bulletin 81-02, CPNPP has modified the PORV block valves by changing the operator gear ratio to guarantee adequate thrust capability and by rewiring the operator for limit closing control.

Additional block valve tests are not necessary because:

1. The operability of block valves is not a safety issue. Plant procedures provide methods for safely shutting down in the event of a small break LOCA.

2. Results of block valve tests performed at the Marshall Station have provided sufficient information to address valve operability.

3. The CPNPP PORV was demonstrated to perform well over a wide range of operating and accident conditions during the EPRI relief valve testing.

4. The probability of a relief valve failure to perform its intended function is low and on the same magnitude as other [[b]] break LOCA initiators. Operability of the associated block valve does not increase the probability of a small break LOCA.

C. ATWS Testing - Although not specifically addressed by the valve testing program, the results provide most of the information necessary to address ATWS events (i.e., relief capability at high pressure).

D. For additional information on CPNPP response to this item, see References 8 and 9.

References

1) EPRI NP-2292-LD; Valve Selection/Justification Report, March 1982

2) EPRI NP-2514-LD; EPRI-Marshall Electric Motor- Operated Valve (Block Valve) Interim Test Data Report, July 1982

3) EPRI NP-2296-LD; Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse-Designed Plants, March 1982

4) EPRI NP-2460-LD; EPRI PWR Safety and Relief Valve Program Test Condition Justification Report, June 1982

5) EPRI NP-2628-LD; Safety and Relief Valve Test Report, September 1982

6) Continuum Dynamics, Quasi-Steady Backpressure For Pressurized Water Reactor Safety and Relief Valves, Vol. III

7) Westinghouse Nuclear Energy Systems, WCAP-10105, Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program, June 1982

8) TXX-4849 dated June 13, 1986, TU Electric response to NRC Request for Additional Information (RAI) dated July 5, 1985.

9) TXX-6398 dated April 15, 1987, TU Electric response to NRC Request for Additional Information (RAI) dated March 27, 1987.

II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

Action Plan Requirements:

“Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.”

- NUREG 0737

CPNPP Response

The pressurizer PORVs have both open and close limit switches with control indicating lights mounted at their respective control switches on the main control board. The limit switches are snap acting, positive throw switches mounted on the valve yokes. They are operated by the actual valve stem motion.

The pressurizer safety valves will be equipped with Reed type valve position indicating switches with indicating lights on the main control boards. Opening of either pressurizer PORVs or safety valves will alarm in the Control Room.

Procedures will be developed/revised to address the RCS relief valve and safety valve indications. The use of this equipment will be integrated into the operator training. This will be implemented by fuel load.

These valve position indicators shall be seismically and environmentally qualified in accordance with IEEE Stds 323-1974 and 344-1975. The indicators shall be qualified and installed prior to fuel load.

TABLE II.D.1-1
SAFETY VALVE INFORMATION

(Sheet 1 of 2)

Safety Valve Information	
-Valve Parameters	
Number of Valves	3
Manufacturer	Crosby Valve & Gage Company
Type	HB-BP-86
Size	6M6
Rated Capacity (steam)	420,000 lbm/hr @2500 psia set pressure
Ring Settings	See Ref. 8
-Inlet Piping Parameters	
Diameter	6 in.
Length (water)	61 in.
Volume (water)	1282 in.3 (5.5 gallon)
Type	Loop Seal-Pressurizer 650°F[(a)]
	-Valve inlet 300°F
-Actuation Transient Parameters	
Fluid Range	Pressurizer-Saturated Steam
	Valve - Subcooled Water
Maximum Back Pressure	500 psi
Relief Valve Information	
-Valve Parameters	
Number of Valves	2
Manufacturer	Copes-Vulcan
Type	Globe D-100-160
Size	3” NPS, 316 W/Stellite Plug and 17-4PH Cage
Capacity (steam)	210,000 lbm/hr @2350 psia
Operator	Copes-Vulcan, Diaphragm Actuating (160 in^2), Reverse Acting, 100 psig air

TABLE II.D.1-1
SAFETY VALVE INFORMATION

(Sheet 2 of 2)

-Inlet Piping Parameters	
Type	Water Seal
-Actuation Transient Parameters	
Fluid range	Pressurizer-Steam
	Relief Valve-Subcooled Water
Block Valve Information	
Manufacturer	Westinghouse
Type	Gate
Model	3GM88
Size	3 in.
Operator	Limitorque SB-00-15

a) Following installation of loop seal insulation

II.E SYSTEM DESIGN

II.E.1 AUXILIARY FEEDWATER SYSTEM

OBJECTIVE:

“Improve the reliability of the auxiliary feedwater system (AFWS).”

- NUREG 0660, Pg. II.E.1-1

II.E.1.1 Auxiliary Feedwater System Evaluation

Action Plan Requirements:

“The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFWS) systems for all PWR operating plant licensees and operating license applications. This action includes:

1. Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages:

2. Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan

3. Reevaluate the AFW system flowrate design bases and criteria.”

- NUREG 0737

4. Based on the analyses performed modify the AFWS, as necessary.

- NUREG 0694, Pg. 22

Also see NRC Letter From D. F. Ross, Jr., dated March 10, 1980.

CPNPP Response

1. The Applicant has performed a simplified Auxiliary Feedwater system reliability analysis similar in method to that described in Appendix III to NUREG-0611. The evaluation is a non-nuclear safety-related study, and is not subject to Quality Assurance. The results of this analysis presented below were prepared in response to the NRC Letter dated March 10, 1980.

   A. System Description

      A complete description of the Auxiliary Feedwater System is in Section 10.4.9. The Auxiliary Feedwater System is designed to provide a supply of high-pressure feedwater to the secondary side of the steam generators for reactor coolant heat

removal following a loss of normal feedwater. The system is comprised of two electric motor-driven pumps and a third turbine-driven pump. All three pumps draw suction from the Nuclear Safety Class 3 Condensate Storage Tanks. A single line supplies water through a common locked-open valve to the suction of the motor-driven pumps, and a second line supplies water to the suction of the turbine-driven pump. The motor-driven pumps normally feed two steam generators each. The turbine-driven pump discharge line branches into four separate lines each feeding one steam generator. The actuation of the Auxiliary Feedwater System following an accident is automatic. A simplified system flow diagram is shown in Figure II.E.1.1-1.

B. Testing

The Auxiliary Feedwater System testing is controlled by the Surveillance Requirements in the CPNPP Technical Specifications.

C. Technical Specification

See Testing above.

D. Reliability Evaluation Results

The results of the reliability evaluations are presented below for the following three events: loss of main feedwater (MFW) with offsite power available, loss of MWF and loss of offsite AC, and loss of feedwater and loss of all AC.

i. Loss of MFW with Offsite Power Available

The simplified fault tree for this transient is shown in Figure II.E.1.1-2 (Sheet 1). The reliability analysis of the Auxiliary Feedwater System based on this event did not identify any single failures that would result in insufficient auxiliary feedwater flow. One double failure was identified. This failure is the result of the closure or blocking of each of the valves in the two supply lines from the condensate storage tank. This double failure is the dominant failure mode.

ii. Loss of MFW and Loss of Offsite AC

The simplified fault tree for this transient is shown in Figure II.E.1.1-2 (Sheet 2). This accident differs from the above only In that the power for the motor driven pumps will come from the diesel generators. To determine what effect this has on overall Auxiliary Feedwater System reliability, the quantified fault tree was expressed as a function of diesel generator reliability. As expected, if the diesels have high reliability, the system reliability is the same as the “loss of feed” event and if the diesels have no reliability, the system reliability is the same as the “loss of feed and the loss of all AC” in the event offsite power is lost. For a diesel generator failure probability as high as .03 the dominant failure mode remains the same as the “loss of feed” event.

iii. Loss of Feedwater and Loss of All AC

The simplified fault tree for this transient is shown in Figure II.E.1.1-2 (Sheet 3). If all AC power is lost, there is only one steam-driven Auxiliary Feedwater pump for this train. The reliability analysis indicates that the dominant failure mode is the steam driven train being out for test or maintenance.

E. Principal Dependencies Identified

The principal dependency found in this analysis was the failure of all trains due to closure of the manual valves in the two supply lines from the condensate storage tank. These valves are locked open and will have special administrative controls-.

There were no design deficiencies noted by this simplified reliability analysis; therefore, no design changes were required.

F. Recommendations

The NRC March 10, 1980 letter instructed applicants to factor the recommendations of Appendix III to NUREG—0611 into plant design. Those recommendations applicable to Comanche Peak are discussed here.

<u>i. GS-1 Tech. Spec. LCO Train Outage Time Limit</u>

CPNPP will have the time limit recommended in the Westinghouse Standard Technical Specifications.

<u>ii. GS—4 Emergency Procedures Backup Water Supply</u>

CPNPP will have an emergency procedure for operators to switch to the backup water supply. See Section 10.4.9.

<u>iii. GS-6 Flow Path Verification</u>

CPNPP will confirm flow path availability of a train that has been out of service to perform periodic testing or maintenance.

<u>iv. Primary AFW Water Source Low Level Alarm</u>

The CPNPP Condensate Storage Tanks have a useable volume of 28,780 gallons of water at the Lo—Low level alarm. Assuming the largest capacity AFW pump is operating, this alarm will allow the operator at least 20 minutes to anticipate the need to make up water or transfer AFW pump suction to the Service Water System.

<u>v. AFW Pump Endurance Test</u>

CPNPP will perform a pump endurance test as part of the pre-startup testing program.

2. For a discussion of how the Auxiliary Feedwater System meets acceptance criteria see FSAR Subsection 10.4.9.

3. Below are listed the AFW System Flowrate design bases and criteria, by question as listed in Enclosure 2 to the March 10, 1980 NRC letter which originally requested the information.

   Question 1

   a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

      1) Loss of Main Feed (LMFW)

      2) LMFW w/loss of offsite AC power

      3) LMFW w/loss of onsite and offsite AC power

      4) Plant cooldown

      5) Turbine trip with and without bypass

      6) Main steam isolation valve closure

      7) Main feed line break

      8) Main steam line break

      9) Small break LOCA

      10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above, The acceptance criteria should address plant limits such as:

      1) Maximum RCS pressure (PORV or safety valve actuation)

      2) Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)

      3) RCS cooling rate limit to avoid excessive coolant shrinkage

      4) Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to Question 1.a

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for Comanche Peak.

- Loss of Main Feedwater Transients
    - Loss of main feedwater with offsite power available
    - Loss of non-emergency AC power (i.e., loss of main 78 feedwater without offsite power available)
- Secondary System Pipe Ruptures
    - Feedline rupture
    - Steamline rupture
- Loss of all AC Power
- Loss of Coolant Accident (LOCA)
- Cool down

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system
- Loss of offsite power with the consequential shutdown of the system pumps, auxiliaries, and controls

The transients are discussed in Sections 15.2.7 and 15.2.6.

The Loss of Non-Emergency AC Power (15.2.6), Loss of Normal Feedwater (15.2.7), and Feedline Rupture (15.2.8) events serve as the basis for the minimum flow required by the Auxiliary Feedwater System for CPNPP. The auxiliary feedwater pumps are sized such that sufficient flow against the steam generator safety valve set pressure (with 3% accumulation) will be provided to remove the stored and residual heat even with the worst single failure in the Auxiliary Feedwater System. The Loss of Non-Emergency AC Power and Loss of Normal Feedwater analyses presented in Chapter 15 demonstrate that the Auxiliary Feedwater System is capable of removing the stored and residual heat, thus preventing the overpressurization of the RCS and loss of water through either the pressurizer relief or safety valves. The Feedline Rupture analyses also demonstrate that the Auxiliary Feedwater System capacity is sufficient to prevent RCS overpressurization and uncovering the reactor core.

Secondary System Pipe Ruptures

The feedwater line rupture accident is discussed in Section 15.2.8.

The main steamline rupture accident is discussed in Section 15.1.5. Auxiliary feedwater is not needed during the early phase of the transient but flow to the faulted loop will contribute to an excessive release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limit on auxiliary feedwater flow delivered to a faulted loop. Eventually, however, the Reactor Coolant System will heat up again and auxiliary feedwater flow will be required to be delivered to the non-faulted loops, but at somewhat lower rates than for the loss of feedwater transients described previously. As discussed in Section 10.4.9., provisions have been made in the design of the Auxiliary Feedwater System to limit, control, or terminate the auxiliary feedwater flow to the faulted loop as necessary in order to prevent containment overpressurization following a steamline break inside containment, and to ensure the minimum flow to the remaining unfaulted loops.

Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only onsite and offsite AC power is lost but also AC emergency power is lost as an assumed common mode failure.

Battery power for operation of protection circuits is assumed available. The Auxiliary Feedwater System provides both an auxiliary feedwater pump power and control source which are not dependent on AC power and which are capable of maintaining the plant at hot shutdown until AC power is restored.

Loss-of-Coolant Accident LOCA

The loss of coolant accidents discussed in Section 15.6.5 do not impose any flow requirements on the AFWS in addition to those required by the other accidents. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the Auxiliary Feedwater System in this transient,

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCA's is basically the same as the system function during hot shutdown or following a spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The Auxiliary Feedwater System may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function of the Auxiliary Feedwater System is discussed in Section 10.4.9.

Response to Question 1.b

Table II.E.1.1-2 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a, above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2.

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability following reactor trip and to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are evaluated by the analysis of the rupture of a main steam pipe transient. The maximum flow requirement determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

a. Maximum reactor power (including instrument error allowance) the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time between initiating eveit End introduction AFWS flow steam generator(s)

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out of 4?

h. RC flow condition – continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate temperature and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory,

Response to 2

Analyses have been performed for the limiting transients which define the AFWS performance requirements. These analyses have been provided for review. Specifically, they include:

- Loss of main feedwater
- Loss of non-emergency AC power
- Rupture of a main feedwater pipe
- Rupture of a main steam pipe inside containment

In addition to the above analyses, calculations have been performed specifically for Comanche Peak to determine the plant cooldown flow (storage capacity) requirements. The loss of all AC power is evaluated via a comparison to the transient results of a Blackout, assuming an available auxiliary pump having a diverse (non-AC) power supply. The LOCA analysis, as discussed in the response Question 1.b, incorporates the system flow requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Main Feedwater/Loss of Non-Emergency AC Power

Analyses of a loss of main feedwater, with and without assuming a loss of power to the reactor coolant pumps, were performed in Sections 15.2.6 and 15.2.7 for the purpose of showing that the peak RCS pressure remains below the criterion for Condition II transients and no water relief occurs through the pressurizer relief or safety valves. Sections 15.2.6 and 15.2.7 summarize the assumptions used in these analyses. The analyses assume that the plant is initially operating at 102% (calorimetric error) of the engineered safety features (ESF) design rating, a conservative assumption in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on low-low steam generator water level, allowing for level 78 uncertainty. Sections 15.2.6 and 15.2.7 show that there is considerable margin with respect to filling the pressurizer even with the most limiting single active failure in the Auxiliary Feedwater System.

These analyses contribute in establishing the capacity of the auxiliary feedwater pumps and also establish train association of equipment so that these analyses remain valid assuming the most limiting single failure.

Rupture of Main Feedwater Pipe

The double ended rupture of a main feedwater pipe downstream of the main feedwater line check valve is analyzed in Section 15.2.8 which summarizes the assumptions used in this analysis. Reactor trip is assumed to occur when the faulted steam generator is at the low-low level setpoint (adjusted for errors). This conservative assumption maximizes the stored heat prior to reactor trip and minimizes the ability of the steam generator to remove heat from the RCS following reactor trip due to a conservatively small total steam generator inventory. As in the loss of normal feedwater analysis, the initial power rating was assumed to be 102% of the ESF design rating. Auxiliary feedwater flow of 430 gpm was assumed to be delivered to the two non-faulted steam generators 1 minute after reactor trip. At 30 minutes after reactor trip, it is assumed that the operator has isolated the AFWS from the break and a flow of 265 gpm from the second motor-driven auxiliary feed pump is delivered to the third non-faulted steam generator. The criterion that the reactor core remains covered with water is met.

This analysis contributes in establishing the capacity of the auxiliary feedwater pumps, establishes requirements for layout to preclude indefinite loss of auxiliary feedwater to the postulated break, and establishes train association requirements for equipment so that the AFWS can deliver the minimum flow required in 1 minute assuming the worst single failure.

Rupture of a Main Steam Pipe Inside Containment

Because the steamline break transient is a cooldown, the AFWS is not needed to remove heat in the short term. Furthermore, addition of excessive auxiliary feedwater to the faulted steam generator will affect the peak containment pressure following a steamline break inside containment. This transient is performed at four power levels for several break sizes. Auxiliary feedwater is assumed to initiate at the time of the break, independent of system actuation signals. The maximum flow is used for this analysis. Table II.E.1.1-3 summaries the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFWS from the faulted steam generator which subsequently blows down to ambient pressure. The criteria stated in Table II.E.1.1-2 are met.

This transient establishes the maximum allowable auxiliary feedwater flow rate to a single faulted steam generator assuming all pumps operating, establishes the basis for runout protection, if needed, and establishes layout requirements so that the flow requirements may be met considering the worst single failure.

Plant Cooldown

Maximum and minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tank size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in the response to Question 1.a, the auxiliary feedwater system partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table II.E.1.1-3 shows the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system.

The cooldown is assumed to commence at the maximum rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to

be removed by the AFWS. See Table II.E.1.1-4 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.

Question 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure, Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response to 3

The Auxiliary Feedwater System (AFWS) is designed, and the equipment specified, considering all criteria established by the NSSS supplier.

Each unit of the CPNPP has two motor driven auxiliary feedwater pumps, and one turbine-driven auxiliary feedwater pump. The turbine driven pump can function without AC power.

FSAR Section 10.4.9 provides the auxiliary feedwater pump capacities.

Pump seals are designed to minimize leakage and expected leakage is considered negligible. The pumps are subjected to periodic inspection, testing and maintenance programs. Performance standards will be set for the pumps. If performance degrades below set limits due to wear, the wear rings in the pump will be replaced. The AFWS is equipped with flow-limiting orifices that will limit the flow to a faulted steam generator from the motor-driven pumps to 700 gpm and from the turbine-driven pump to 600 gpm. The total flow to the faulted steam generator will not be more than 1300 gpm, which is less than the 1380 gpm maximum flow stated in Table II.E.1.1-3. The other steam generators will receive more than the minimum required flow.

AFWS flow rates are provided in Sections 15.1.5, 15.2.6, 15.2.7 and 15.2.8. The flow rates assumed in the accident analyses have considered the most limiting single active failure.

4. See Response to 1.E above.

II.E.1.2 Auxiliary Feedwater System Automatic Initiation And Flow Indication

Action Plan Requirements:

"a. Automatic Initiation of the Auxiliary Feedwater System

Provide automatic initiation of all auxiliary feedwater systems, The initiation signals and circuits shall be designed in such a manner that a single failure will not result in the loss of auxiliary feedwater system function. Testability of the initiating signals and circuits shall be a feature of the design. The initiating signals and circuits shall be powered from the emergency buses. Manual capability to initiate the auxiliary feedwater system from the control room must be retained and must be implemented in such a manner that a single failure in the manual circuits will not result in the loss of system function. The a-c motor driven pumps and valves in the auxiliary feedwater system must be included in the

automatic actuation (simultaneous or sequential) of the loads to the emergency buses. The design of the automatic initiating signals and circuits must be such that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room."

NUREG 0578, Pg. 10

"b. Auxiliary Feedwater Flow Indication to Steam Generators

Provide safety-grade indication in the control room of auxiliary feedwater flow for each steam generator. The flow instrument channels shall be powered from the emergency buses, consistent with satisfying the power diversity requirements for auxiliary feedwater systems."

- NUREG 0578, Pg. 11

Also see NUREG 0737.

Response to a.

Auxiliary feedwater system initiation as described herein is safety-related. The CPNPP design complies with the stated positions of NUREG-0578 also as described in the detailed responses below; therefore, no design changes were required as a result of related Action Plan requirements. Components which provide this function are listed in Table 17A-1.

NUREG—O578 Positions 1 to 7

"Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:"

NUREG—O578 Position 1

"The design shall provide for the automatic initiation of the auxiliary feedwater system."

CPNPP Response

The Auxiliary Feedwater System consists of three separate trains. Two of the trains each consist of one 50 percent capacity electric motor driven auxiliary feedwater pump, valves, piping and controls, The third train consists of a 100% capacity turbine driven auxiliary feedwater pump, valves, piping and controls. See FSAR Fig. 10.4-11. The electric motor driven pumps start automatically on:

1. Two out of four Lo-Lo steam generator level signals from any one of four steam generators

2. Trip of both main feedwater pumps (as sensed by hydraulic pressure switches on the feedwater pumps)

3. Station blackout sequence signal

4. Safety injection signal.

The turbine driven pump starts automatically on:

1. Two out of four Lo-Lo steam generator level signals from any two of four steam

2. Station blackout sequence signal.

The pneumatic flow control valve in each of the auxiliary feedwater lines to the steam generators is a normally open, fail open valve. The motor-operated and manual isolation valves in these lines are maintained open by administrative procedure.

<u>NUREG-0578 Position 2</u>

“The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.”

<u>CPNPP Response</u>

As shown in our response to NUREG-0578, Position 1 above, the auxiliary feedwater initiation circuitry is part of the Engineered Safety Features (ESF) system, and as such, is installed in accordance with IEEE Standard 279. This Standard is referenced in 10 CFR 50.55a(h).

<u>NUREG-0578 Position 3</u>

“Testability of the initiating signals and circuits shall be a feature of the design.”

<u>CPNPP Response</u>

The auxiliary feedwater initiation signals and circuitry are testable. Such testability is included as a requirement of the technical specifications.

<u>NUREG-0578 Position 4</u>

“The initiating signals and circuits shall be powered from the emergency buses,”

<u>CPNPP Response</u>

As shown in our responses to Positions 1 and 2, the initiating signals and circuits are a part of the plant engineered safety features. The requirements for the ESF system dictate that the system shall meet the “single failure criteria.” To accomplish this, the initiating signals and circuits are powered from separate emergency buses.

The initiating sensors such as steam generator low-low level are powered from separate and redundant nuclear instrumentation and control panels, each of which is supplied by either on-site emergency generators or station emergency batteries. Each of the two redundant Solid State Protection System (SSPS) trains is supplied by separate power sources.

NUREG-0578 Position 5

“Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.”

CPNPP Response

Manual initiation for each train exists in the control room. No single failure in the manual initiation portion of the circuit can result in the loss of auxiliary feedwater system function.

NUREG-0578 Position 6

“The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.”

CPNPP Response

As with all ESF equipment, the a-c motor-driven pumps and all valves in the system are automatically transferred to, and sequentially loaded on, the emergency buses on loss of offsite power.

NUREG-0578 Position 7

“The automatic initiating signals and circuits shall be designed so that their failures will not result in the loss of manual capability to initiate the AFWS from the control room.”

CPNPP Response

All automatic and manual initiating signals and circuits are installed in accordance with regulatory requirements and are safety grade and redundant. No single failure in the automatic portion of the system will result in loss of the capability to manually initiate the AFWS from the control room.

NUREG-0578 Position

“In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety grade requirements.”

CPNPP Response

As described in the preceding responses, the automatic initiating signals presently meet safety grade requirements. No upgrading is required.

CPNPP Response to b.

Auxiliary feedwater system flow as described herein is safety-related. The CPNPP design complies with the stated positions of NUREG-0578 also as described in the responses below; therefore, no design changes were required as a result of related Action Plan requirements. Components which provide this function are listed in Table 17A-1.

NUREG-0578 Positions 1 and 2

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the Control Room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

NUREG-0578 Position 1

Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the Control Room.

CPNPP Response

Two safety-grade indicators of auxiliary feedwater flow to each steam generator are provided in the Control Room. In addition, safety-grade auxiliary feedwater flow indication is provided for each auxiliary feedwater pump.

These flow indicators are included in the human factors analysis of the Control Room which is discussed under item I.D.1.

NUREG-0578 Position 2

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of Standard Review Plan, Section 10.4.9.

CPNPP Response

Of the two safety-grade indicators of auxiliary feedwater flow to each steam generator, one is powered from a Train A vital instrument ESF bus and the other from a Train B vital instrument ESF bus.

The pump flow indicators are powered from the vital instrument ESF bus of the same train as the pump. The turbine driven pump flow indication is from Train A.

II.E.3 DECAY HEAT REMOVAL

OBJECTIVE:

“Improve the reliability and capability of nuclear power plant systems for removing decay heat and achieving safe shutdown conditions following transients and under post accident conditions.”

- NUREG 0600, Pg. II.E.3-1

II.E.3.1 Reliability of Power Supplies for Natural Circulation

Action Plan Requirements:

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

\- NUREG 0737

CPNPP Response

Power is supplied to four pressurizer heater groups from offsite power, when available, and from the onsite emergency diesel generators through ESF buses. Redundancy is provided by supplying two groups of pressurizer heaters from each redundant ESF train, Control power for manual on/off control of each of these four heater groups is supplied from the 125 volt DC ESF bus in the same train as the main power supply.

Procedures are established to control Reactor Coolant System pressure and temperature. Per Westinghouse analysis, pressurizer heaters are not required during natural circulation cooldown to Hot Shutdown or Cold Shutdown.

No loads need to be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

The electrical interface between each of the four pressurizer heater groups and its associated F bus of concern is through a circuit breaker which trips on an “S” signal and is qualified in accordance with safety-grade requirements. The manual controls for these breakers are qualified in accordance with safety-grade requirements.

II.E.4 CONTAINMENT DESIGN

OBJECTIVE:

“Improve the reliability and capability of nuclear power plant containment structures to reduce the radiological consequences and risks to the public from design basis events and degraded—core and core—melt accidents.”

- NUREG 0660, Pg. II.E.4-1

II.E.4.1 Dedicated Penetration

Action Plan Requirements:

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single—failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

- NUREG 0737

CPNPP Response

Post accident combustible gas contol vis external hydrogen purge or internal electric hydrogen recombiners is no longer required as described in Section 6.2.5. See Section 6.2.4 for containment penetration isolation.

II.E.4.2 Isolation Dependability

Action Plan Requirements:

“Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signal.”

- NUREG 0578, Pg. 8

"(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4, (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

"(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be nonessential, describe the

basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

"(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

"(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

"(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

"(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed, as defined in SRP 6.2.4, Item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days." (A copy of the Staff Interim Position is enclosed as Attachment 1.)

"(7) Containment purge and vent isolation valves must close on a high radiation signal.

- NUREG 0737

<u>CPNPP Response</u>

1. There are two phases of containment isolation at CPNPP. Phase A isolates all nonessential process lines but does not effect safety injection, containment spray, component cooling water supplied to the reactor coolant pumps, or auxiliary feedwater. Phase B isolates all process lines except safety injection, containment spray, and auxiliary feedwater.

   Phase A isolation is initiated by high containment pressure, low steam line pressure, low pressurizer pressure, or manual initiation. Phase B isolation is initiated by high-high containment pressure or manual initiation.

   In addition, containment ventilation isolation is initiated by Phase A isolation, containment spray actuation, or high containment radioactivity.

   This system fully complies with Section 11.6 of the SRP 6.2.4. Compliance with all remaining sections of SRP 6.2.4 is documented in Section 6.2.4.

2. A reevaluation of the classification assigned to each process penetration has been performed and is described in revised Section 6.2.4.3.

3. All nonessential systems use either normally closed manual valves or the valves are automatically isolated on a containment isolation signal. See revised Section 6.2.4.3.

4. The CPNPP design complies with the position. The operator must reset each device individually as desired. See revised Section 7.3.1.1.4.

5. The CPNPP containment pressure setpoint (Hi-1) that initiates containment isolation for non-essential penetrations is 3.2 psig. This setpoint is the minimum compatible with normal operating conditions and the value used in accident analyses.

6. The containment isolation valves for the containment purge system are automatic isolation valves. Furthermore, these valves will be closed during operational modes 1, 2, 3 and 4 (see response to Question 022.20). These valves will be verified to be closed at least every 31 days.

7. CPNPP complies as described in Section 6.2.4.

TABLE II.E.1.1-1
PRIMARY EVENT PROBABILITIES

Figure II.E.1.1-2 Variables	Event	Value
P_a	Blocked Butterfly Valve 006	1×10^{-4} (a)
P_b	Blocked Butterfly Valve 007 Using Coupling	1×10^{-1}
P_c	Train A Logic Failure Using Coupling	7×10^{-3} (a)
P_d	Train B Logic Failure Using Coupling	.2
P_e	Turbine Train Logic Failure	.5
P_f	No Manual Start	5×10^{-3} (a)
P_g	Train A pump mechanical failure	1×10^{-3} (a)
P_h	Train A pump control circuit failure	4×10^{-3} (a)
P_i	Turbine Pump Mechanical failure	1×10^{-3} (a)
P_j	No Steam to the Turbine Train	1×10^{-3}
P_k	Train Out for test or Maintenance	7.8×10^{-3} (a)
P_l	Diesel Generator Failure is assigned a variable probability	
P_m	Manual Valve Blocked	1×10^{-4} (a)
P_n	Check Valve Blocked	1×10^{-4} (a)
P_o	Air Operated Valve Blocked	3×10^{-4} (a)
P_p	Motor Operated Valve blocked	1×10^{-4} (a)
P_r	Not Used	

a) Values taken from NUREG-0611, Table III-2

TABLE II.E.1.1-2
CRITERIA FOR AUXILIARY FEEDWATER SYSTEM DESIGN BASIS CONDITIONS

Condition or Transient	Classification[a]	Criteria	Additional Design Criteria
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure +10%. No consequential fuel failures	Pressurizer does not fill.
Loss of Non-Emergency AC Power	Condition II	(Same as LMFW)	Pressurizer does not fill.
Feedline Rupture	Condition IV	10CFR100 dose limits. Containment design pressure not exceeded	Core does not uncover.
Loss of all A/C Power	N/A	Note 1	Same as loss of non-emergency A/C power assuming turbine driven pump
Loss of Coolant	Condition IIII	10 CFR 100 dose limits 10 CFR 50 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 50 PCT limits	
Main Steamline Break	Condition IV	Containment design pressure and temperature not exceeded	
Cooldown	N/A		100°F/hr 557° to 350°F

a) Ref: ANSI N18.2 (This information provided for those transients performed in the FSAR).

Note:

1 Although this transient establishes the basis for AFW pump powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.

TABLE II.E.1.1-3
SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

(Sheet 1 of 2)

Transient		Cooldown	Main Steamline Break (containment)
a.	Max reactor power	3644 MWt	0, 30, 70, 102% of rated (percent of 3425 MWt)
b.	Time delay from event to Rx trip	2 sec	variable
c.	AFWS actuation signal/time delay for AFWS flow	N/A	Assumed immediately @ 0 sec (no delay)
d.	SG water level at time of reactor trip	N/A	N/A
e.	Initial SG inventory	70,760 lbm/SG at 544.6°F	consistent with power
	Rate of change before & after AFSW actuation	N/A	N/A
	Decay heat	N/A	ANS + 20%
f.	AFW pump design pressure	1236 psia	N/A
g.	Minimum # of SGs which must receive AFW flow	N/A	N/A
h.	RC pump status	Tripped	All operating
i.	Maximum AFW temperature	100°F	440°F
j.	Operator action	N/A	10 minutes
k.	MFW purge volume/ S/G temperature	450 ft^3/440°F	800 ft^3/loop (for dryout time)

TABLE II.E.1.1-3
SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

(Sheet 2 of 2)

Transient		Cooldown	Main Steamline Break (containment)
l.	Normal blowdown	none assumed	none assumed
m.	Sensible heat	Table II.E.1.1-4	N/A
n.	Time at standby/time to cooldown to RHR	2 hr/4 hrs	N/A
o.	AFW flow rate	variable	1380 gpm (constant) to faulted SG.

Note:

Refer to Sections 15.2.6, 15.2.7 and 15.2.8 for the assumptions used in the Loss of Non-Emergency AC Power, Loss of Main Feedwater and Feedline Break events.

TABLE II.E.1.1-4
SUMMARY OF SENSIBLE HEAT SOURCES

Primary Water Sources (initially at rated power temperature and inventory)

- RCS fluid
- Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at rated power temperature)

- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at rated power temperature and inventory)

- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping.

Secondary Metal Sources (initially at rated power temperature)

- All steam generator metal above tube sheet, excluding tubes.

TABLE II.E.1.1-5
THIS TABLE HAS BEEN DELETED

II.F INSTRUMENTATION AND CONTROLS

OBJECTIVE:

"Provide instrumentation to monitor plant variables and systems during and following an accident. Indications of plant variables and status of systems important to safety are required by the plant operator (licensee) during accident situations to (1) provide information needed to permit the operator to take preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered safety features systems, and manually initiated systems are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity); (3) provide information to the operator that will enable him to determine the potential for a breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and if a barrier has been breached; (4) furnish data for deciding on the need to take unplanned action if an automatic or manually initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation; (5) allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of the impending threat; and (6) improve requirements and guidance for classifying nuclear power plant instrumentation, control, and electrical equipment important to safety."

-NUREG 0660, pg. II.F-1

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION.

Action Plan Requirements:

"Item II.F.1 of NUREG-0660 contains the following subparts:

(1) Noble gas effluent radiological monitor:

(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position);

(3) Containment high-range radiation monitor;

(4) Containment pressure monitor;

(5) Containment water level monitor; and

(6) Containment hydrogen concentration monitor.

"NUREG-0578 provided the basic requirements associated with items (1) through (3) above. Letters issued to all operating nuclear power plants dated September 13, 1979 and October' 30, 1979 provided clarification of staff requirements associated with items (1) through (6) above. Attachments 1 through 6 present the NRC position on these matters."

-NUREG-0737

CPNPP Response

The use of the additional accident monitoring instrumentation as listed will be integrated into the operating procedures and training programs prior to fuel load:

(1) Noble Gas Monitor

The CPNPP design will include wide range noble gas monitors for the plant vent stack which will detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident.

An adjacent-to-line monitor will be provided for each main steam line to monitor the concentration in steam that may be released to the environment by the safety or relief valves.

For description of these monitors, see Section 11.5.

(2) Iodine/Particulate Sampling

The wide range noble gas monitor discussed above provides the capability to sample the plant vent stacks as required.

For description, see Section 11.5.

(3) Containment High Range Radiation Monitor

The redundant Category 1 monitors will be located in each Containment Building at Elevation 905’-9”. Exact location is provided in Figure II.F-1. To ensure valid data, these monitors will be located at least 90E apart and will not be located adjacent to process piping.

Monitor special calibration and environmental qualification will be performed as specified in Table II.F.1-3 of NUREG-0737.

For further discussion, see Section 12.3.

(4) Containment Pressure

The CPNPP design will include redundant wide range pressure indication (0 - 150 psig) on the Main Control Board meeting Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display.

The present CPNPP design includes four channels of intermediate range pressure indication (-5 to + 60 psig) on the Main Control Board meeting Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display.

(5) Containment Water Level

CPNPP design will include redundant wide range containment level indication (elevation 808’-3” to 817’ - 6) on the Main Control Board meeting the intent of Regulatory Guide 1.97 Rev. 2 requirements. In addition this parameter will be provided as input to the SPDS high level display. These transmitters measure volume in excess of 600,000 gallons. CPNPP design will also include normal sump level indications (0 - 3 feet) on the Main Control Board. The containment wide range level indication covers the maximum expected flooding levels assuming the reactor cavity is blocked and does not receive any flood water to simulate the most limiting post-accident condition (flooding starts at El. 808’). Actual post-accident conditions would include flooding into the reactor cavity sump (781’-2”) and reactor cavity (783’-7”). Therefore, containment narrow range level indication is not considered as required for accident monitoring.

(6) Containment Hydrogen

See Sections 6.2.5.2.3 and 7.5 for containment hydrogen monitoring.

II.F.2 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

Action Plan Requirements:

“Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.”

-NUREG-0737

CPNPP Response

The CPNPP design will include redundant instrumentation to monitor the approach to, existence of and recovery from inadequate core cooling. The monitored parameters will be the reactor coolant system (RCS) saturation margin, the collapsed water level above the reactor core and the RCS temperature at the core exit.

An indication of a declining margin to saturation in the RCS will provide the earliest warning that conditions are developing which could lead to ICC. If the event is allowed to progress, saturation conditions will be observed along with an indication of a declining collapsed water level above the reactor core. Collapsed water level alone does not identify the existence of ICC, only the potential for ICC. Maintaining the collapsed water level at a point above the core is not essential for adequate core cooling. A steam/water froth region extending down into the core could equate to a collapsed water level below the top of the core and yet provide adequate core cooling. Only as the top of the froth region drops below the top of the core would ICC tend to occur. RCS pressure and core exit temperatures would indicate this phase of the event by a “saturation” margin in the superheat region. Alternatively, the recovery from ICC and the subsequent stages

of the event can be monitored to verify that corrective actions taken have resulted in the expected plant response.

The instrumentation to be used for monitoring the RCS saturation margin, collapsed water level, and core exit temperatures is described below.

II.F.2.1 RCS Saturation Margin

The core cooling monitor is a microprocessor-based system capable of determining the saturation margin in the RCS. This determination is based on the lower measurement of the hot leg or pressurizer pressure and the highest measured temperature in the RCS. The temperature used for the most current calculation and the saturation margin are displayed together on the main control board. Redundant microprocessor systems and sensor devices will maintain independence between the two separate trains of temperature and pressure data.

II.F.2.2 Collapsed Water Level

To provide the capability for measurement of the reactor coolant inventory in the upper head and plenum regions of the reactor vessel, CPNPP has decided to install a heated junction thermocouple (HJTC) system supplied by Combustion Engineering, Inc. (C-E). This system includes two identical probes located 180° apart, each in the immediate vicinity of a cold leg inlet. Each probe assembly contains a series of eight HJTC sensors with individual splash shields which are axially distributed inside a separator tube. An HJTC sensor consists of two physically separated thermocouple junctions, one of which is electrically heated. The basic principle of the system is to determine whether or not a sensor is covered with water by detecting the temperature difference between adjacent heated and unheated thermocouples.

The HJTC probe has undergone extensive tests by C-E and has demonstrated its ability to indicate the coolant inventory in the reactor vessel above the core.

When the collapsed water level inside the probe falls below a given sensor location during a loss-of coolant event, the heated junction temperature increases due to the relatively poor cooling ability (lower heat transfer coefficient) of steam instead of water. When the relative temperature difference between heated and unheated junctions exceeds a predetermined value, the sensor registers as being uncovered (i.e., surrounded by steam only). Once a sensor becomes uncovered, the potential increase in temperature of the heated junction depends upon the sensor heater power and the pressure. At low pressures, a high heater power can produce excessive junction temperatures and possible sensor failure. Therefore, a heater power control system is used to preclude excessive temperatures. However, should sensor damage occur due to overheating, the loss of a single HJTC sensor within a probe will not seriously compromise the resolution of usefulness of the HJTC probe measurements due to system redundancy and the independence of the sensors within the probe.

The probes for CPNPP will be of the “split-probe” design, having two sensors located in the upper head region and six sensors located in the upper plenum region. Sudden depressurization of the upper plenum could result in localized voiding with no immediate comparable occurrence in the upper head region due to the limited pressure equalization vent area in the upper support plate. The resulting pressure disequilibrium could result in high ambiguous indications without the “split-probe” design. This design will allow an unambiguous indication of collapsed water level in either region regardless of their instantaneous relative pressure. The measurement of

collapsed water level in either region above the core will give immediate indication to the reactor operator that a loss-of-coolant condition exists and that the approach to ICC is imminent unless proper corrective actions are taken.

The control board display will be in the form of two “mimic” probes, each composed of a column of light emitting diodes (LED’s) corresponding to the eight HJTC sensors in each probe. Sensors which are surrounded by water will be indicated by energizing the corresponding LED.

II.F.2.3 Reactor Coolant System Temperatures

The original, vendor supplied system of core exit thermocouples (CET) and resistance temperature detectors (RTD) will be used to supply the necessary temperature data for the core cooling monitoring system. The saturation margin monitor will be the primary data link for the CET’s. The plant process/ERF computers will have on-line access to the data but are secondary and isolated from this primary link. The core cooling monitor will pass a complete set of input and output data in engineering units to these computers following each calculation cycle. The set of fifty CET’s have been divided into two redundant trains, each representative of all four quadrants of the core. Although originally a non-safety class system, necessary modifications have been made to upgrade all of the system components to Class 1E. Maximum achievable train separation is maintained; however, complete separation is impossible in the reactor head region. Due to retrofit limitations, the two trains share reactor vessel head penetrations where a degree of physical separation is achieved with each individual thermocouple in a separate stainless steel sleeve. Above the vessel head, the messenger wires of the two safety trains are bundled separately and are physically separated from each other and from non-safety Train C cables to the greatest extent possible. Complete train separation is achieved above the CRDM Seismic Support Assembly’s platform (Unit 1) and the missile shield (Unit 2). All connectors and cold junction terminals have been upgraded to Class 1E. Thermocouple leads from the thermocouples to the cold junction terminals are mineral-oxide insulated and stainless steel jacketed.

The average bulk coolant temperature in each hot or cold leg of the primary reactor coolant loop is individually monitored by three RTDs in separate thermowells in each leg near the reactor vessel. These are also divided into two trains with either a hot or cold leg from each of the four loops represented in each train.

Computer software for the core cooling monitor is designed to limit the error introduced by the calculations themselves to less than 0.5°F. Thus, the primary sources of system inaccuracy are sensor inaccuracy first and the initial analog to digital conversion second. Core Cooling Monitor system accuracy is ±3.5°F. User modifiable parameter ranges are incorporated in the software to facilitate automatic testing for out-of-range data. The primary thermocouple temperature limits directly result from these ranges and from the potential loss of physical integrity of Type K thermocouples at elevated temperatures. Instrumentation ranges are listed in Table II.F.2-1.

II.F.2.4 Implementation

CPNPP will establish acceptable operation of the upgraded CETs and the core cooling monitors prior to fuel loading in Unit 1. Necessary modifications to the RCS pressure boundary have been incorporated to allow installation of the HJTC probes. Final installation and implementation of the HJTC probes are complete.

CPNPP Emergency Operating Procedures (EOPs) have incorporated the saturation margin monitor instrumentation and the HJTC instrumentation. The CPNPP EOPs are based on the analyses and generic Emergency Response Guideline Procedures developed by the Westinghouse Owners Group.

TABLE II.F.2-1
SATURATION MARGIN MONITOR AND CORE EXIT THERMOCOUPLE
SPECIFICATION SUMMARY

	PARAMETER	RANGE
INPUT:	Coolant Pressure (1)	0-3000 PSIG
	Pressurizer Pressure (1)	1700-2500 PSIG
	Coolant Temperature (4)	0-700°F
	Type K Thermocouple (25)	0-2300°F
	Cold Junction Temperature (3)	32°-500°F
OUTPUT:	Margin (1)	300°F subcooling to 300°F superheat
	Hottest Thermocouple (1)	0-2300°F
	Data (2)	All inputs and outputs in engineering units each cycle.

Core Cooling Monitor System Accuracy: ± 3.5°F

II.G ELECTRICAL POWER

OBJECTIVE:

"Increase the reliability and diversification of the electrical power supplies for certain safety-related equipment."

-NUREG 0660, pg. II.G-1

II.G.1 POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES AND LEVEL INDICATORS

Action Plan Requirements:

"Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available."

-NUREG 0737

CPNPP Response

The Pressurizer PORVs are pneumatically operated valves with pilot solenoids.

The pilot solenoids are powered from 125 volt DC emergency buses which can be powered from batteries, offsite power or emergency onsite power.

The Pressurizer PORV controls are powered from the inverter buses which are capable of being fed from offsite and onsite emergency ac power, and the onsite dc power system.

The Pressurizer PORV block valves for CPNPP are motor operated valves. The power for the motors is supplied from class 1E 480 volt buses which may be supplied from either offsite power or onsite emergency diesel generators. Control power is provided from the same buses.

The Pressurizer PORV dc power and control power and the PORV block valve motor power and control power are connected to different emergency ESF buses through devices that are qualified in accordance with safety-grade requirements.

The Pressurizer level indicated circuits are safety-grade and post-accident qualified. AC power for these Class 1E instrument channels is supplied from inverters which are supplied from ESF buses with automatic backup from the emergency batteries.

The power supplies for the PORVs and block valves are listed appropriately in Table 17A-1.

II.K MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENT AND LOSS-OF-FEEDWATER ACCIDENTS

OBJECTIVE:

"To perform systems reliability analyses and to effect changes in emergency operating procedures and operator training to improve the capability of plants to mitigate the consequences of the small-break loss-of-coolant accidents (LOCA) and loss-of-feedwater events."

- NUREG 0660, Pg. II.K-1

II.K.1 IE BULLETINS

Action Plan Requirements:

"C.1.5* Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning. (See Bulletin 79-06A Item 8, 79-06B Item 7, 79-08 Item 6.)"

"C.1.10 Review and modify, as required, procedures for removing safety-related systems for service (and restoring to service) to assure operability status is known. (See Bulletin 79-05A Item 10, 79-06A Item 10, 79-06B Item 9, 79-08 Item 8.)"

"C.1.17 For Westinghouse-designed reactors, trip the pressurizer low-level coincident signal bistables, so that safety injection would be initiated when the pressurizer low-pressure setpoint is reached regardless of the pressurizer level. (See Bulletin 79-06A and Revision 1, Item 3.)"

-NUREG 0694, pg. 17

CPNPP Response

II.K.1 Procedures will be reviewed and, as necessary, revised to address the removal of safety-related systems from service (and restoring to service) to assure proper operability.

II.K.1.5 CPNPP will review all safety-related valve positions, positioning requirements, and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features.

CPNPP will review and modify as necessary those related maintenance, testing, startup, and supervisory periodic surveillance procedures to ensure that safety-related valves are returned to their correct positions following manipulations and are so maintained during all operational modes.

II.K.1.10 CPNPP will review and modify as necessary the maintenance and test procedures to ensure that they require:

(a) Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

(b) Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

(c) Explicit notification of involved reactor operations personnel whenever a safety-related system is removed from and returned to service.

II.K.1.17 FSAR Section 15.6 discusses CPNPP protection from decrease in reactor coolant inventory. Pressurizer low level trips are not utilized at CPNPP.

II.K.2.13 Thermal Mechanical Report – Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident With No Auxiliary Feedwater

Action Plan Requirements:

“A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.”

Changes to Previous Requirements and Guidance

“Also, this requirement has been changed to include all operating pressurized-water reactors (PWRs) and applicants.”

- NUREG 0737

CPNPP Response

II.K.2.13 To address the requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small-break LOCAs with loss of all feedwater, the Westinghouse Owner’s Group (WOG) has initiated a number of programs applicable to generic Westinghouse plants. Results of these programs are summarized in the following reports.

(1) “Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants,” WCAP-10019, December 1981 (transmitted via Ref. #1).

(2) “Summary of Evaluations Related to Reactor Vessel Integrity performed for the Westinghouse Owner’s Group,” May 1982 (transmitted via Ref. 2).

Although these reports are primarily directed to demonstrating that there are no near-term safety concerns for operating Westinghouse plants, the

analytical methodologies and general conclusions are applicable to CPNPP.

Pressurized thermal shock is not expected to be a problem at CPNPP because of reactor vessel material properties. Based on the conservative material assumptions described in the Technical Specifications, the limiting material RT_{ndt} changes from an initial value of 40°F to 110°F (1/4 T) after 16 EFPY of fast neutron fluence is well below the proposed NRC screening criterion of 270°F (Ref. 3).

The limiting RT_{ndt} will be benchmarked and updated during CPNPP operation using neutron fluence data provided by the in-situ capsule surveillance program. This updating will include the reevaluation of the prediction of RT_{ndt}. If an extrapolated RT_{ndt} value were to violate the NRC screening criteria, then CPNPP would respond with an appropriate action plan according to NRC requirements.

Reference

1. Letter OG-66, dated December 30, 1981, O. D. Kingsley (WOG) to H. R. Denton (NRC).

2. Letter OG-70, dated May 28, 1982, O. D. Kingsley (WOG) to H. R. Denton (NRC).

3. “NRC Evaluation of Pressurized Thermal Shock” (draft), U.S. NRC, September 13, 1982.

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Action Plan Requirements:

“Analyze the potential for voiding the reactor coolant system (RCS) during anticipated transients.”

Changes to Previous Requirements and Guidance

“The previous requirement has been to include all PWR operating reactors and applicants.”

- NUREG 0737

CPNPP Response

II.K.2.17 Westinghouse (in support of the Westinghouse Owner’s Group) has performed a study which addresses the potential for void formation in Westinghouse designed Nuclear Steam Supply Systems during natural circulation cooldown/ depressurization transients. The study has been submitted to the NRC by the Westinghouse Owners Group (Letter OG-57, dated April 20, 1981, R. W. Jurgensen to P. S. Check) and is applicable to Comanche Peak Nuclear Power Plant.

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (Letter OG-64, dated November 30, 1981, R. W. Jurgensen to D. G. Eisenhut). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific consideration) will be utilized in the implementation of the CPNPP Emergency Operating Procedures.

II.K.2.19 Sequential Auxiliary Feedwater Flow Analysis

Action Plan Requirements:

“Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.”

Changes to Previous Requirements and Guidance

“The previous requirement has been changed to include all operating pressurized-water reactors (PWRs) and applicants for operating license.”

- NUREG 0737

CPNPP Response

II.K.2.19 This requirement is not applicable to plants with inverted U-tube steam generators such as the Westinghouse steam generators for Comanche Peak.

II.K.3 FINAL RECOMMENDATIONS FOR B&O TASK FORCE

II.K.3.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System

Action Plan Requirements:

“All PWR licensees should provide a system which uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.”

“Each licensee should perform a confirmatory test of the automatic block valve closure system following installation.”

- NUREG 0737

CPNPP Response

I.K.3.1 Based on the reduction in PORV LOCA frequency due to post-TMI modifications already implemented, and automatic PORV block valve closure system is unnecessary and therefore not incorporated. See the response to II.K.3.2 below and WCAP-9804 for further discussion.

II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

Action Plan Requirements:

"(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck- open power-operated relief valve (PORV) and show how these actions constitute sufficient improvements in reactor safety."

"(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above."

- NUREG 0737

CPNPP Response

II.K.3.2 The Westinghouse Owners Group submitted WCAP-9804, "Probabilistic Analysis and Operational Data in Response to Item II.K.3.2 for Westinghouse NSSS Plants" to the NRC on March 13, 1981. This report describes various modifications to Westinghouse plants since TMI and, using probabilistic analysis via event trees, estimates the effect of the post-TMI changes, the modifications described in WCAP-9804 have been incorporated by CPNPP. Therefore, the improvements in reactor safety demonstrated by the WCAP are applicable.

II.K.3.3 Reporting of SV & RV Failures & Challenges

Action Plan Requirements:

"Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report."

- NUREG 0694, Pg. 24

CPNPP Response

II.K.3.3 Reporting of SV and RV Failures and Challenges

The CPNPP Emergency Plan requires prompt notification to the NRC of the failure of a SV or RV to close. The reporting of SV & RV challenges was moved from the annual report to the monthly operating report (TS 6.9.1.5), then License Amendment 64 deleted this reporting requirement altogether.

II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-Of-Coolant Accident

Action Plan Requirements:

"Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623."

- NUREG 0737

CPNPP Response

II.K.3.5 TMI Action Plan Item II.K.3.5 has been evaluated in accordance with the guidelines set forth in NRC Generic Letter 83-10c. The Westinghouse Owner's Group (WOG) has performed a series of analyses to determine (1) what unique indicators exist that would distinguish between a SBLOCA and a SGTR, and (2) the 10 CFR 50, Appendix K and Best Estimate predictions of the Peak Cladding Temperatures (PCT) associated with the delayed Reactor Coolant Pump (RCP) trip scenario.

Reference 1 provided the basis for calculating the three (3) unique distinguishing identifiers that exist for CPNPP. The generic WOG Emergency Response Guidelines were then used to develop the plant specific Emergency Operating Procedure (EOP) for the scenario. The CPNPP EOPs will require a manual trip of the RCPs when conditions warrant the action. Reference 2 documents that even for the most limiting Westinghouse plant, the PCTs are below the limits established in NRC Generic Letter 83-10c. This result provides the justification for allowing a manual trip of the RCPs.

References:

(1) OG-110, dated 1 December 1983, J. J. Sheppard (WOG) to R. J. Mattson (NRC).

(2) OG-117, dated 12 March 1984, J. J. Sheppard (WOG) to R. J. Mattson (NRC).

II.K.3.9 Proportional Integral Derivative Controller Modification

Action Plan Requirements:

"The Westinghouse-recommended modification to the proportional integral derivative (PID) controller should be implemented by affected licensees."

- NUREG 0737

CPNPP Response

II.K.3.9 CPNPP has reconfigured the pressurizer power operated relief valve (PORV) controller for proportional-integral (PI) control action. Without derivative action the contribution to the error signal from reactor coolant pressure rate of change has been eliminated. This results in fewer challenges to the reactor protection system from spurious pressure signal noise.

II.K.3.10 Proposed Anticipatory Trip Modification

Action Plan Requirements:

“The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck- open power-operated relief valve (PORV) is substantially unaffected by the modification.”

- NUREG 0737

CPNPP Response

II.K.3.10 The CPNPP design includes the P-9 interlock which blocks the direct reactor trip on a turbine trip at or below 50% of rated thermal power. Analyses have been performed which demonstrate that a turbine trip without direct reactor trip at or below 50% of rated thermal power does not pose any undue or additional challenges to the pressurizer PORVs. Assuming all control systems are operable, neither the RCS PORVs nor the steam generator ARVs reached their respective pressure setpoints. Even when single credible failures in the control systems are considered, the pressurizer PORVs do not reach their setpoints. Also see FSAR Section 7.2.

II.K.3.11 Justification of Use of Certain PORVs

Action Plan Requirements:

“Demonstrate that the PORV installed in the plant has a failure rate that is not significantly less than the valves for which there is an operating history.”

- NUREG 0694, Pg. 18

CPNPP Response

II.K.3.11 The CPNPP Pressurizer Power Operated Relief Valves are Copes Vulcan Model No. D-100-160 globe valves. The report on failure rates required by II.K.3.2 will establish the operating history for these valves.

II.K.3.12 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip

Action Plan Requirements:

"Licensees with Westinghouse-designed operating plants should confirm that their plants have an anticipatory reactor trip upon turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for one installation of this trip."

- NUREG 0737

CPNPP Response

II.K.3.12 CPNPP has the anticipatory reactor trip on turbine trip as described in Section 7.2.1.1.2, item 6.

II.K.3.17 Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes

Action Plan Requirements:

"Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation)."

Changes to Previous Requirements and Guidance

"This clarification adds the requirement to propose changes that will improve and control availability."

- NUREG 0737

CPNPP Response

II.K.3.17 ECCS Outages

The NRC has closed this item for CPNPP. The requirements of 10 CFR 50.72 and 50.73 and reporting on the NPRDS is adequate for reporting ECCS outages.

II.K.3.25 Effect of Loss of Alternating-Current Power on Pump Seals

Action Plan Requirements:

"The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The

pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.”

Changes to Previous Requirements and Guidance

“The evaluation and proposed modifications shall be submitted by July 1, 1981. The May 7, 1980 letter called for modifications by January 1, 1982. This clarification adds a documentation requirement for the evaluation to the submitted by July 1, 1981. The modification date remains unchanged. Additionally, this task has changed to include Westinghouse and Combustion Engineering operating reactors and operating reactor applicants.”

- NUREG 0737

CPNPP Response

II.K.3.25 Loss of cooling water to the reactor coolant pump seal coolers due to loss of offsite power is not applicable to CPNPP because the Component Cooling Water supply to the Non-safeguards loop of the CCWS is through motor operated butterfly valves (1-HV-4526 and 4527 at location E-3 and 1-HV-4524 and 4525 at location D-3 on FSAR Figure 9.2-3) which fail-as-is on loss of power. As discussed in Section 5.4.1.3.1 the effect of loss of offsite power is to cause a temporary stoppage of the cooling water. Power is restored to the component cooling water pumps after 30 seconds (see Table 8.3-2). Therefore, cooling of the seal water heat exchanger and the reactor coolant pump packages described in FSAR Section 9.2.2.2 is restored in a timely manner on a loss of offsite power. The components described are covered under the appropriate items in Table 17A-1.

II.K.3.30 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance With 10 CFR Part 50, Appendix K

Action Plan Requirements:

“The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.”

Changes to Previous Requirements and Guidance

“The changed requirement (1) allows for justification of acceptability of present small-break LOCA models by comparison with test date, and (2) requests each licensee to outline scope and schedule for model revision or comparison with test data by late fall, 1980. The original requirement did not allow provision for showing acceptability of present models by comparison with plant data.”

- NUREG 0737

CPNPP Response

II.K.3.30 TUGCO has participated, as a member of the Westinghouse Owners Group in the development of the modified SBLOCA code NOTRUMP. A technical description of the code was submitted to the NRC via References 1 and 2. A favorable SER was released on May 21, 1985, Reference 3.

References

(1) Westinghouse Letter, E. P. Rahe to C. O. Thomas, NS-EPR-2694, December 22, 1982. (Transmitting WCAP- 10054)

(2) Westinghouse Letter, E. P. Rahe to C. O. Thomas, NS-2681, November 12, 1982. (Transmitting WCAP-10079)

(3) Safety Evaluation. TMI Action Item II.K.3.30 for Westinghouse Plants attached to NRC letter V. S. Noonan to M. D. Spence dated June 6, 1985.

II.K.3.31 Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46

Action Plan Requirements:

“Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.”

- NUREG 0737

CPNPP Response

II.K.3.31 In Reference 1, the NRC Staff indicated that the resolution of TMI Action Plan Item II.K.3.31 may be accomplished by generic analyses to demonstrate that the previous NRC approved WFLASH SBLOCA results were conservative when compared with the new NOTRUMP SBLOCA. Such generic studies were undertaken by the Westinghouse Owners Group (WOG) of which Texas Utilities is a participating member. The WOG completed these generic studies and has submitted the results of the analyses to the NRC in the topical report WCAP-11145 (Reference 2). Topical report WCAP-11145 is therefore referenced in order to demonstrate that the requirements to TMI Action Plan II.K.3.31 are satisfied for CPNPP in a generic fashion in accordance with Reference 1.

Topical report WCAP-11145 documents the results of a series of Small Break LOCA (SBLOCA) analyses performed with the NRC approved NOTRUMP SBLOCA Evaluation Model. Cold leg break spectrum analyses were performed for the limiting SBLOCA plant form each of the Westinghouse 4- loop, 4-loop Upper Head Injection (UHI), 3-loop, and 2- loop plant categories. The limiting SBLOCA plant in each category was defined on the basis of previous SBLOCA analyses which were performed with the WFLASH SBLOCA code. In addition to

the cold leg break spectrums, a hot leg and pump suction break were performed as part of the 4- loop plant analyses, confirming that the cold leg was still the worst break location. Comparison of the NOTRUMP cold leg break spectrum results with the previously generated WFLASH results, showed that the WFLASH results were conservative for all plant categories. In particular, for the previous bounding 4-loop peak clad temperature (PCT) plant the NOTRUMP SBLOCA code calculated a PCT which was 537 °F lower than the WFLASH code calculated value. This result is consistent with the other generic studies which indicated that the NOTRUMP SBLOCA calculated PCTs tended to be significantly lower than the previously bounding WFLASH PCTs.

The generic results documented in WCAP-11145, demonstrate that a plant specific reanalysis of CPNPP with the NOTRUMP SBLOCA should result in the calculation of a limiting PCT which would be significantly lower than the 1788°F PCT currently calculated with the WFLASH. Hence, the WFLASH results which currently form the licensing basis for CPNPP are conservative and still valid for demonstrating the adequacy of the Emergency Core Cooling System to mitigate the consequences of a SBLOCA, as required by 10CFR50.46. It is therefore concluded that a plant specific analysis is not needed in order for CPNPP to comply with TMI Action Item II.K.3.31. Rather, Texas Utilities references WCAP-11145 in order to comply with TMI Action Item II.K.3.31 on a generic basis in accordance with Reference 1.

References

1) NRC Generic Letter 83-35 from D. G. Eisenhut, “Clarification of TMI Action Plan Item II.K.3.31”, November 2, 1983.

2) L. D. Butterfield letter to J. Lyons, “Westinghouse Owner’s Group Transmittal of WCAP-11145”, OG-190, June 11, 1986.

III EMERGENCY PREPAREDNESS AND RADIATION EFFECTS

III.A EMERGENCY PREPAREDNESS AND RADIATION EFFECTS

III.A.1 IMPROVE LICENSEE EMERGENCY PREPAREDNESS - SHORT TERM

OBJECTIVE:

"Promptly improve and upgrade licensee emergency preparedness by requiring improvements in facilities, plans, procedures, offsite support, technical assistance, equipment and supplies required to adequately respond to and manage an accident."

- NUREG 0660, Pg. III.A.1-1

III.A.1.1 Upgrade Emergency Preparedness

Action Plan Requirements:

"Licensees will upgrade emergency preparedness in accordance with the requirements described in the NRC "Action Plan for Promptly Improving Emergency Preparedness" (SECY 79-450), which was distributed to all licensees during regional meetings in August 1979, and in accordance with subsequently issued acceptance criteria (NUREG-0654). These actions include:

(1) Preparing and submitting upgraded plans which satisfy the NRR supplemental acceptance criteria provided by the NRC emergency preparedness review teams, with special attention to the establishment of emergency action levels in accordance with NUREG-0610, "Basis for Emergency Action Levels for Nuclear Power Facilities."

(2) Implementing the short-term emergency planning recommendations of NUREG-0578.

(3) Establishing an onsite Technical Support Center, an onsite Operational Support Center, and a near-site Emergency Operations Facility.

(4) Establishing improved offsite radiological monitoring capability in accordance with the NRR/RAB technical position.

(5) Providing planning assistance to appropriate Federal, State, and Local governments to assure that their emergency response roles are properly coordinated with the facility plan and that such plans satisfy the NRC acceptance criteria.

(6) Providing resources as necessary to State and Local governments for implementing the emergency planning zone concept, in accordance with NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."

(7) Participating in periodic joint exercises involving Federal, State, and Local government emergency response organizations.

- NUREG 0660, Pg. III.A.1-9

"Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparedness and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations."

- NUREG 0694, Pg. 25

<u>CPNPP Response</u>

1) The CPNPP Emergency Plan has been written in accordance with the draft version of NUREG-0654 and was submitted to the NRC in October 1980 for review.

2) For implementation of NUREG-0578, 2.1.8.a, Post Accident Sampling, see FSAR Section II.B.3.

   For implementation of NUREG-0578, 2.1.8.b, Accident Monitoring Instrumentation, see FSAR Section II.F.1.

   For implementation of NUREG-0578, 2.1.8.c, Improved Inplant Iodine Instrumentation, see FSAR Section III.D.3.3.

3) See FSAR Section III.A.1.2 and CPNPP Emergency Plan, Section 6.0.

4) See CPNPP Emergency Plan, Section 7.0.

5) Planning assistance is provided to State and Federal agencies upon request. Local planning assistance is provided by Luminant Power.

6) Resources to implement the State and Local emergency plans will be made available to those agencies.

7) The State and Local emergency plans provide for periodic joint exercises with CPNPP.

III.A.1.2 Upgrade Licensee Emergency Support Facilities

<u>Action Plan Requirements:</u>

"A separate technical support center shall be provided for use by plant management, technical, and engineering support personnel. In an emergency, this center shall be used for assessment of plant status and potential offsite impact in support of the control room command and control function. The center should also be used in conjunction with implementation of onsite and offsite emergency plans, including communications with an offsite emergency response center. Provide at the onsite technical support center the as-built drawings of general plant arrangements and

piping, instrumentation, and electrical systems. Photographs of as-built system layouts and locations may be an acceptable method of satisfying some of these needs."

"Each operating nuclear power plant should establish and maintain a separate onsite operational support center outside the control room. In the event of an emergency, shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room shall report to this center for further orders and assignment."

- NUREG 0578, Pg. 13

Also see NRC Letter, dated April 25, 1980; August 1, 1980; and September 5, 1980.

CPNPP Response

I. INTRODUCTION

CPNPP established emergency response facilities (ERFs), consisting of a Technical Support Center (TSC), an Operational Support Center (OSC), and an Emergency Operations Facility (EOF), prior to issuance of an operating license. These facilities meet the guidelines of NUREG 0578, 0654, 0694, and also meet the intent of NUREG 0696 "Functional Criteria for Emergency Response Facilities," as stated in the description below.

The ERFs have been upgraded by incorporation of an integrated data acquistion and display system. This system provides data to the TSC and EOF and also provides the Safety Parameter Display System (SPDS) displays to the Control Room, TSC and EOF. A description of this system is provided in the discussion below.

The CPNPP Emergency Plan has been revised to describe the existence and function of the current CPNPP emergency response facilities; therefore, facility description in Sections IV, V and VI below for the TSC, OSC and EOF, respectively, are not described in FSAR Section III.A.1.2 but are relocated to the CPNPP Emergency Plan.

II. EMERGENCY RESPONSE FACILITY INSTRUMENTATION, DATA SYSTEM EQUIPMENT, AND POWER SUPPLIES

The data display equipment provided in the TSC and EOF is part of the integrated Emergency Response Facility (ERF) Computer System. As shown in Figure III.A.1.2-2, the integrated ERF Computer System gathers, stores and displays data needed in the TSC and EOF to analyze the plant conditions. The system performs its function independent of action in the Control Room and without degradation or interfering with Control Room and plant functions. Two of three TSC displays are powered by a Non-1E battery backed, Uninterruptable Power Supply (UPS) System; one TSC display is powered from a Non-Class 1E 120 VAC common supply; EOF displays are powered from an AC source (the 25 KV Plant Support Power Loop).

Signals to the integrated ERF Computer System, which are received from sources providing signals to safety-related equipment or displays, are isolated to ensure that the system will not degrade performance of the safety-related equipment or displays.

Power supplied to the ERF computer will be provided from a Non-1E Battery UPS System to insure the continuity of TSC functions if temporary loss of primary TSC Power sources occurs. Upon loss of all A/C power or a seismic event, the ERF computer will not be functional; however, data acquisition remains available from other sources. Should power fail, no data will be lost, as the displays can be regenerated when the power is restored.

The design goal for the data system reliability is to achieve an operational unavailability goal of 0.01 during all plant operating conditions above cold shutdown.

The SPDS display equipment used in the TSC and EOF is not seismically qualified; it will have the same design goal of meeting the TSC and EOF data system reliability and performance criteria. The design of the TSC and EOF data system equipment incorporates human factors engineering with consideration for both operating and maintenance personnel. The TSC and EOF display equipment discribed herein is included under the appropriate items in Table 17A-1.

III. TECHNICAL DATA AND DATA SYSTEM

The ERF Computer System receives, stores, processes, and displays information acquired from different areas of the plant as needed to perform the TSC and EOF function. The ERF Computer System provides access to accurate and reliable information sufficient to determine:

1. Plant steady-state operating conditions prior to an accident.

2. Transient conditions producing the initiating event.

3. Plant systems dynamic behavior throughout the course of an accident.

The data set available to the data system is complete enough to permit accurate assessment of an accident without interference with the Control Room emergency operation.

The data set available to the ERF Computer System includes Category 1 and applicable Category 2 Type A, B, C and E Accident Monitoring variables (as described in FSAR Section 7.5) and Type D Accident Monitoring variables which are also identified as Type D variables in Regulatory Guide 1.97, Rev. 2, Table 2.

All sensor data and calculated variables used in the data set for SPDS, EOF, or for transmission to offsite locations are available for display. The accuracy of the data displayed is substantially the same as the accuracy of comparable data displayed in the Control Room. The time resolution of data acquisition is sufficient to provide data without loss of information during transient conditions where that data is necessary for operator action. The time resolution of each sensor signal will respond on the potential transient behavior of the variable being measured.

Disk and tape storage and recall capability are provided for the TSC and EOF data set. Two hours of pre-event and 12 hours of post event data are recorded.

The sample frequency is consistent with the use of the data. Capacity to record two weeks of additional post event data are provided by the ERF Computer System. Archival data storage is provided automatically, and retrieval is accomplished without interrupting the TSC and EOF data acquisition function. The ERF displays (3 of which are provided in the TSC and 2 in the EOF), if used for data retrieval, will not be available for real time parameters, but can be returned to on line display service very quickly.

A sufficient number of display and printout devices is provided in the TSC and EOF to allow designated personnel to perform their assigned tasks. The displays include alphanumeric and graphical representations of:

1. Plant system variables

2. In-plant radiological variables

3. Meteorological information

Trend information and time history display capability are available to personnel in the TSC and EOF. The displays are designed so that call up, manipulation, and presentation of data can easily be performed. The data display presents information in easily understood formats.

The SPDS can also be displayed in the TSC and EOF.

IV. TECHNICAL SUPPORT CENTER (TSC)

See the CPNPP Emergency Plan, Section 6 and specifically 6.2.

V. OPERATIONAL SUPPORT CENTER (OSC)

See the CPNPP Emergency Plan, Section 6 and specifically 6.3.

VI. EMERGENCY OPERATIONS FACILITY (EOF)

See the CPNPP Emergency Plan, Section 6 and specifically 6.4.

VII. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

A. FUNCTION

The purpose of the Safety Parameters Display System (SPDS) is to assist control room personnel in evaluating the safety status of the plant. The SPDS provides a continuous high level graphical display of plant parameters or derived variables representative of the safety status of the plant.

As an operator aid, SPDS serves to concentrate a minimum set of plant parameters from which the plant safety status can be assessed. More detailed plant information will be provided by several secondary displays.

Human factor engineering is incorporated in the SPDS design to enhance the functional effectiveness of the control room personnel.

The SPDS will be operational during normal, as well as, emergency conditions except for loss of all A/C power or seismic events. However, data acquisition will remain available from other sources. The SPDS will be able to display pertinent information concerning steady-state and transient conditions.

All portions of the SPDS have been designed and specified such that all design basis safety questions have been considered.

Revision 3 to the SPDS Safety Analysis Report contains further supporting information and analysis, and was transmitted via Reference [1].

B. LOCATION

SPDS displays are available in the Control Room, TSC, and EOF. The SPDS location in the control room will be visible to both the control room operator and the Senior Reactor Operator.

C. SIZE

The SPDS will be compatible with the existing space in the control room area and will not interfere with normal movement or with full visual access to other control room operating systems or displays.

D. STAFFING

The SPDS will be of such design that no operating personnel, in addition to the normal control room operating staff, are required for its operation.

E. DISPLAY CONSIDERATIONS

Luminant Power was a member of the Safety Assessment System committee to develop and implement the necessary displays to meet the requirements for the SPDS. The committee consisted of Quadrex Corporation and eleven other utilities owning Westinghouse PWRs. The Luminant Power implementation of the generic SAS design has been integrated into the Emergency Response Facility Computer System (ERFCS). The following discussion serves as a general description of the SPDS portion of the ERFCS.

1. General Considerations

The ERFCS satisfies the requirements of the Safety Parameter Display System (SPDS). This section describes the portion of the ERFCS which meets the SPDS requirements of NUREG-0696 and the intent of NUREG-0737 Supplement 1. The ERFCS provides a centralized, flexible, computer-based data and display system to assist control room personnel in evaluating the safety status of the plant. This assistance is accomplished by providing the operator and also Emergency Response Facilities (ERFs) a high-level graphical display containing a minimum

set of key plant parameters representative of the plant safety status. More detailed plant information is provided by several secondary displays. All graphical displays are presented to the control room operator on high resolution multiple color monitors.

All data displayed by the SPDS is validated by comparing redundant sensors, checking the value against reasonable limits, calculating rates of change, and/or checking temperature versus pressure curves.

All displays on the SPDS have been carefully designed by persons with plant operating experience and evaluated against human factors design criteria. The intent of the SPDS is to present to the control room personnel a few easily understandable displays which use color coding and pattern recognition techniques to indicate off-normal values.

These displays are updated and validated on an essentially real-time basis.

The SPDS will be operable during normal and abnormal plant operating conditions. The SPDS will operate during all SPDS required modes of plant operation. The normal operation mode will encompass all plant conditions at or above normal operating pressure and temperature. When the reactor coolant system is intentionally cooled below normal operating values, the operator will select the Heatup-Cooldown mode which alters the limit checking algorithm for the key parameters. An additional mode may be provided to address concerns of cold shutdown plant conditions.

2. Display Hardware Locations and Operation

The SPDS portion of the ERFCS may be implemented on a single CRT located in a control location of the control room visible to the control room operator and the Senior Reactor Operator. This CRT contains the high-level display from which the overall safety status of the plant may be assessed. A dedicated function button panel allows operator selection of several predetermined second level (trend) displays at any time.

The SPDS has been designed such that control room personnel can utilize its features without requiring additional operations personnel.

The SPDS displays have been provided to other ERFs, such as the Technical Support Center and Emergency Operations Facility.

3. Display Contents

The primary display consists of bar graphs of selected parameter values, digital status indicators for important safety system parameters and digital values. The parameters indicated by bar graphs and digital values include: RCS pressure, RCS temperature, pressurizer level, steam generator levels and steam generator pressures. Status indicators are provided for containment environment, secondary system radiation, and reactor vessel water level. Core exit

temperature, amount of subcooling and containment radiation are indicated by digital values.

In addition, there is a message area which will be used to indicate that an appropriate secondary display provides further information in case an off-normal value is detected or an event is occurring.

Each of the bar graphs indicate wide-range values. If a parameter's value is outside the normal range, the bar color will turn yellow, then red. Arrows next to the bar indicate the trend direction (increasing or decreasing) based on data smoothing algorithms.

Secondary displays may be selected by the operator. Trend graph groups of selected parameters, showing the last thirty minutes of plant operation are available. These trend groupings were chosen to keep like parameters or related parameters on one display "page."

4. Human Factor Engineering

Human factors engineering and industrial design techniques have been effectively combined to establish man-machine interface design requirements, maximize system effectiveness, reduce training and skill demands, and minimize operator error.

The CRT color graphic formats and functional keyboard designs have been developed through an interdisciplinary team of senior operational, human factors, industrial design and computer interface personnel.

Minimum use of color combined with simplified format through the CRT presentation have key design features to provide both normal and off-normal pattern recognition. The operator, who is the end user, has been directly involved from the conception to insure that man-machine interface goals of SPDS have been satisfied. Human factor engineering standards and testing verification have been used which are consistent with accepted practices.

5. Validation and Verification

The SPDS is implemented on a digital computer system which includes a peripheral display generator computer for color graphic displays. The software that controls the sensor data validation, key parameter construction, and display formats has been developed under a software quality assurance program based on IEEE-730. Extensive verification and validation activities have been performed on the system using NSAC-39 as a guideline. Additional verification and validation activities have been performed using ANSI/ANS 10.4-1987 as a guideline.

F. DESIGN CRITERIA

The total SPDS will not be Class 1E or meet the single-failure criterion. The sensors and signal conditioners (such as preamplifiers, isolation devices, etc.) will be designed and

qualified to meet Class 1E standards for those SPDS parameters that are also used by safety systems.

The control room operating staff will be provided with sufficient information and criteria to allow for performance of an operability evaluation of SPDS if an earthquake should occur.

The SPDS, as used in the control room, will be designed to an operational unavailability goal of 0.01. The cold shutdown unavailability goal for the SPDS during the cold shutdown and refueling modes for the reactor will be 0.2.

The unavailability goal of 0.01 is more stringent than can be reasonably achieved without some redundancy. Therefore, dual minicomputers, data multiplexors, and other critical peripherals have been installed.

The isolation of the SPDS from safety-related devices is described as follows for each type of safety-related device that has input to the SPDS computer system (e.g. multiplexed analog and digital inputs, core cooling monitor inputs, radiation monitoring system inputs, reactor vessel level indicaiton system inputs).

1. Multiplexed analog and digital inputs are obtained from various systems throughout the plant (e.g., analog signals from tank levels, system pressures, flow rates, etc., digital signals from valve postion, breaker position, etc.). All multiplexed inputs from safety-related devices are routed to their respective (Train A or B) qualified multiplexer cabinets (MUX) located in the environmentally protected cable spreading room. Since the output of the qualified multiplexers is used as input to the non-qualified SPDS computer system, an appropriate qualified isolation device was chosen that would electrically isolate the MUXs from the SPDS and also allow the rapid transmission of data. The electrical isolation is accomplished with qualified optical isolators. All credible adverse conditions were considered and tested per specifications given in the qualification test reports. The isolation devices are in compliance with applicable NRC guidelines and IEEE standards for environmental and seismic qualification, and the entire MUX system is designed and qualified for fail-safe operation.

2. Inputs to the SPDS from the core exit thermocouples and the margin to saturation indicator are by way of the two (Train A and B) qualified safety-related Core Cooling Monitors (CCMs). Electrical isolation between the CCMs and the SPDS is accomplished by means of dc to dc converters and optical isolators. DC to dc converters provide isolated dc power to the non-Class 1E CCM circuits by isolating the Class 1E power supply from the non-Class 1E circuits. Optical isolators provide electrical isolation between the Class 1E input signal circuitry and the non-Class 1E output signal circuitry. Testing was performed to verify that the input side of the isolator suffered no degradation as a result of faults applied to the isolator output. The application of the maximum credible fault voltage and current meets this criterion, thus assuring the adequacy of the dc to dc converter and optical isolator as an isolation device for this application. Adequacy was determined by vendor testing, which is documented in Reference [2].

3. All inputs to the SPDS from the Radiation Monitoring System (RMS) are transmitted as ASCII data from the RMS computer system, except for

containment radiation inputs which are routed to the multiplexers as described in Item 1. The RMS computers are non-safety-related devices; therefore, no provision for electrical isolation between them and the SPDS is required.

4. All inputs to the SPDS from the qualified Reactor Vessel Level Indication System (RVLIS) are transmitted as ASCII data through an FOC data link from the qualified, safety-related RVLIS cabinet to the SPDS computer system. All credible adverse conditions were considered and tested per specifications given in the qualification test reports. The isolation devices are in compliance with applicable NRC guidelines and IEEE standards for environmental and seismic qualification, and the entire RVLIS system is designed for fail-safe operation.

The power supply will be from Non-1E Battery, UPS supply.

Expandability in both numbers of parameters and processing power will be provided.

Verification and validation are applied as appropriate.

References

1) Letter from W. J. Cahill, Jr. (TUElectric) to the NRC, “Submittal of the SPDS Safety Analysis Report, Revision 3”, TXX-89531, July 31, 1989.

2) Letter from W. J. Cahill, Jr. (TUElectric) to the NRC, “Submittal of Core Cooling Monitor (CCM) Isolator Test Report”, TXX-89079, February 17, 1989.

VIII. INTEGRATED ERF COMPUTER SYSTEM

A. GENERAL DESCRIPTION

In general, the CPNPP integrated ERF computer system consists of a system configuration as shown in Figure III.A.1.2-2. The overall system principally consists of:

1. Data Acquisition System (DAS) units that isolate, digitize, and transmit field sensor signals to the ERF computers.

2. Redundant minicomputers will provide the data processing/distribution/ and record keeping functions required. The minicomputers will be powered from a highly reliable Non-1E Battery/UPS System.

3. For each unit the display system will consist of eight (8) human factored color graphics display units implementing the display generated by the Westinghouse Ad-hoc Safety Assessment System Group. Three displays will be located in the control room, three in the TSC, and two in the EOF. One of the control room displays will be dedicated to the display of SPDS type parameters. An additional CRT display on the Main Control Board is available to display SPDS parameters.

However, this display is also used to display plant process parameters. Selection of the data source for this display is accomplished via a manual switch located on the Main Control Board. The other displays will have full display capability, including the SPDS type parameters, in addition to all other parameters available to the computer. This system is referred to as the display system. The control room, and two of three TSC displays will be powered from Non-1E, Battery/UPS power supplies. One TSC display is powered from a Non-Class 1E 120 VAC common supply.

The integrated ERF Computer System reliability design goal is to achieve 0.01 unavailability during all plant operating modes above cold shutdowns.

B. IMPLEMENTING SCHEDULE

The ERF Computer hardware and software was installed with a sufficient set of input variables to accomplish the SPDS and Regulatory Guide 1.23 functions by fuel load.

III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS--LONG TERM

OBJECTIVE:

To upgrade the emergency preparedness of nuclear power plants.

- NUREG 0660

Action Plan Requirements:

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants.”

- NUREG 0737

CPNPP Response

The CPNPP Emergency Plan is currently being revised to reflect requirements expressed in NUREG-0654, Revision 1. Equipment, facilities and instruments necessary to implement the Emergency Plan are subject to an Operations Quality Assurance Program. Activities necessary to implement the Emergency Plan are controlled by approved procedures and subject to the proper QA controls and surveillances.

Prior to fuel load of Unit One, the meteorological measurements program for CPNPP shall consist of the following:

1) A primary meteorological measurements program;

2) A backup meteorological measurements system;

3) A system for making near real-time predictions of the atmospheric effluent transport and diffusion;

4) A capability for remote interrogation, on demand, of the atmospheric measurements and prediction systems by CPNPP emergency response organizations and the NRC staff.

III.A.3 IMPROVING NRC EMERGENCY PREPAREDNESS

OBJECTIVE:

To enable NRC, in the event of a nuclear accident at a licensed reactor facility, to (1) monitor and evaluate the situation and potential hazards, (2) advise the licensee's operating staff as needed, and (3) in an extreme case, be able to issue orders governing such operations.

III.A.3.3 Communications

Action Plan Requirements

Install direct dedicated telephone lines between each plant and the NRC Operations Center.

- NUREG 0694, Pg. 27

CPNPP Response

The Federal Telecommunications System (FTS) 2000 network has been installed at CPNPP for communications with the NRC Operations Center.

III.D RADIATION PROTECTION

III.D.1 RADIATION SOURCE CONTROL

OBJECTIVE:

"Perform evaluations to establish additional design features that should be included in the rulemaking proceeding of Item II.B.8. The purpose of these evaluations is to identify design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident."

-NUREG 0660, Pg. III.D.1-1

III.D.1.1 Primary Coolant Sources Outside The Containment Structure

Action Plan Requirements:

"Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

"(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests as intervals not to exceed each refueling cycle."

- NUREG 0737

CPNPP Response

The residual heat removal (RHR) system; portions of the containment spray (CS) system; portions of the safety injection (SI) system; portions of the chemical and volume control (CVCS) system and the RCS Sampling System (post accident sampling system portion only until such time as a modification eliminates the PASS penetration as a potential leakage path) have been identified as systems which process primary coolant, and could contain high level radioactive materials. (See Section II.B.2.2.) Programs will be implemented to reduce and maintain leakage to as-low-as-practical levels. These programs will include but not be limited to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

1. The RHR, CS, SI, and CVCS systems are ASME Code Class 2 and 3, and are subject to the in-service inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI, including pressure tests.

2. At intervals not exceeding refueling outages, Radioactive System Leakage Inspection (RSLI) tests will be performed on appropriate portions of the RHR, SI, CS, CVCS and RCS Sampling Systems.

3. Excessive leakage into controlled areas will also be indicated by abnormally high airborne radioactivity levels.

4. Plant Initial Leakage Rates

   The initial leakage rates of the system to be tested will be determined from the system hydrostatic (or pneumatic) test which is conducted prior to plant operations and/or the initial Radioactive System Leakage Inspection (RSLI) test.

5. Test Methods

   The primary testing method will be by system walkdown while at normal operating pressure with quantified measurements obtained at all visually observed leakage paths. Systems or subsystems not readily testable in this manner will require either leakage makeup to maintain test pressure testing or pressure drop testing. Actual testing experience may suggest alternate methods which may be incorporated.

6. Limiting Leakage Value

   The effort at CPNPP will be to minimize the overall leakage of fluid from potentially radioactive systems to the environment. Therefore, the limiting leakage value will be based on a cumulative total of the systems within each Unit. The leakage will be limited to a maximum of 1 gpm/Unit.

7. Maintaining Leakage Value

   While shutdown or during operations, the affected systems will receive a periodic inspection for leakage at intervals not to exceed each refueling cycle. The leak tests will be scheduled to coincide with routine system operability tests whenever possible during operations. To minimize the actual leakage rate, packings or seals with provisions for adjustment will be adjusted whenever leakage is noted during the inspection. Replacements of seals, gaskets, o-rings, etc., will be accomplished on a selective basis to prevent exceeding the limiting leakage value.

   Should the limiting leakage value be exceeded, a reinspection and leak test will be conducted on the system(s) with high leakage rates. Excessive leakage paths will be identified and evaluated. Areas of continued high leakage will be considered for modification or improvement in repair methods.

   Modifications or repairs will be accomplished as quickly as possible. Consideration as to the type of leakage, fluid involved, leakage location and collectability will influence repair or modification schedules.

8. Incidence of Leakage at North Anna

A reviews and comparison of the designs of North Anna and Comanche Peak as pertains to the leakage incidence indicates that the design of the Comanche Peak volume control tank relief system is sufficiently different to preclude a similar occurrence. In addition a further review of procedures and as-built configurations will be performed as they become available. Areas or activities which could result in release of radioactive material will be evaluated for modification.

III.D.3 WORKER RADIATION PROTECTION IMPROVEMENT

OBJECTIVES:

“Improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents, by improving radiation protection plans, health physics, inplant radiation monitoring, control room habitability, and radiation worker exposure data base.”

III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions

Action Plan Requirements:

"(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

"(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.”

- NUREG 0737

CPNPP Response

An in-plant radiation monitoring program shall be established, implemented and maintained to ensure the capability to accurately determine the airborne/iodine concentration in vital areas under accident conditions. This program shall include provisions for the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment.

During accident conditions portable grab sampling equipment will be used to obtain iodine samples. These samples may be taken in all areas occupied by plant personnel.

The portable grab sampling equipment used consists of a high volume air sampler, a sample cartridge head and a silver zeolite iodine cartridge. This equipment is identical to the equipment normally used for routine iodine sampling. The only difference is the type of collection media used (silver zeolite versus charcoal).

After the iodine sample is obtained, the silver zeolite cartridge will be surveyed and counted in accordance with applicable Emergency Plan Procedures. Specific calculations contained in these procedures will allow personnel taking samples to determine if iodines are present. All used silver zeolite cartidges will be properly labeled, bagged and retained for further analysis, if required. Should additional analysis be required, the selected samples will be transported to the Chemistry Hot Lab. In the Hot Lab, the silver zeolite cartridges will be prepared for counting, then transferred to the Counting Room for isotopic analysis.

In the Counting Room, the silver zeolite cartridge will be counted on a Gamma Spectroscopy System consisting of: a HPGe detector, Multi-Channel Analyzer and Process Computer. The process computer identifies the iodine isotopes present, performs the calculations, and prints out a listing of all iodine isotopes identified along with their respective concentrations. If, due to accident conditions, the normal counting room is unavailable or inaccessible for reasons including, but not limited to high background radiation, the EOF-NOSF Laboratory Facilities may be utilized to identify and quantify iodine samples.

During normal operations, the EOF-NOSF laboratory facilities are used for training purposes. These facilities can be available for performing radiological analysis (gross radiation level and isotopic characterization of environmental samples). The NOSF laboratory facilities are not maintained in a constant state of readiness since they are used as training facilities. Several other lab facilities can be made available on short notice to provide back-up radiological analysis of post accident samples. These other facilities are listed in Section 6.8 of the CPNPP Emergency Plan. Availability and location of non-portable radiation detection/measuring equipment is listed in FSAR Table 12.5-1. Both stationary and portable monitors are commercial grade, and subject to an appropriate Operations QA Program.

All procedures and instructions associated with in-plant iodine monitoring under accident conditions shall be prepared, approved, and implemented prior to Unit 1 fuel load.

All training programs and procedures associated with sample acquisition, sample quantification and data evaluation pertaining to in-plant iodine monitoring activities under accident conditions shall be prepared, approved, and implemented prior to Unit 1 fuel load.

III.D.3.4 Control Room Habitability Requirements

Action Plan Requirements:

"In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50)."

- NUREG 0737

CPNPP Response

A reevaluation of control room habitability has been performed.

Sections 2.2.1, 2.2.2, 2.2.3 and 6.4 of the Standard Review Plan (SRP) were reviewed for compliance and are discussed as follows:

1) FSAR Sections 2.2.1 and 2.2.2 is in compliance with SRP Sections 2.2.1 and 2.2.2.

2) FSAR Section 2.2.3 is in compliance with SRP Section 2.2.3.

3) FSAR Section 6.4 shows that the present CPNPP design for the control room is in compliance with SRP Section 6.4. An analysis was performed to evaluate the effect of a postulated steam generator blowdown piping break in the Electrical and Control Building on the Control Room. The results of the analysis indicate that the Control Room is not affected.

Information required by NUREG-0737, III.D.3.4, Attachment 1, is provided below:

(1) The control room ventilation mode of operation for radiological accident isolation is described in FSAR Section 6.4.2.

(2) Control room characteristics

  (a) The control room air volume is provided in FSAR Section 15.6.5.4 (item 4(j)).

  (b) The control room envelope (i.e., emergency zone) is described in FSAR Section 6.4.

  (c) The control room ventilation system schematic is provided by FSAR Figure 9.4-1.

  (d) Control room infiltration leakage is discussed in FSAR Sections 6.4.2.3 and 9.4.1.2.

  (e) HEPA filter and charcoal adsorber efficiencies are provided in FSAR Section 15.6.5.4 (item 4(b)).

  (f) The distance between the containment and air intake is provided in FSAR Section 15.6.5.4 (item 4(g)).

  (g) Layouts are provided in FSAR Figures 1.2-1, 1.2-33, 1.2-34 and 6.4-3.

  (h) Control room shielding is discussed in FSAR Section 6.4.2.5.

  (i) Automatic isolation capability is discussed in FSAR Sections 6.4.2.2 and 9.4.1.2.

  (j) Toxic chemicals are discussed in FSAR Section 2.2.3.

  (k) Portable self-contained breathing apparatus (SCBA) are provided in the control room. There will be an adequate supply of air to sustain the five-man emergency team for a six-hour period. At least one SCBA will be provided for each member of the emergency team. There will be one additional SCBA to serve as a spare in case one unit fails.

(l) Bottled air supplys are discussed in FSAR Section 6.4.1.3.

(m) Emergency food and water supplies are provided in FSAR Section 6.4.1.1.

(n) The Shift Manager controls the number of personnel allowed into the control room during both normal and emergency conditions. The personnel capacity is dependent on specific conditions at the time in question. The minimum capacity under design basis accident conditions is five men for a period of five days.

(o) Potassium iodide (KI) will be available to plant emergency personnel. The details of KI use and availability are controlled in accordance with provisions of the CPNPP Emergency Plan.

(3) Onsite Storage of Chlorine and Other Hazardous Chemicals

Toxic chemicals as discussed in FSAR Section 2.2.3.

(4) Offsite manufacturing, storage, or transportation facilities of hazardous chemicals is discussed in FSAR Section 2.2.

(5) Technical Specifications for CPNPP will comply with the Standard Technical Specifications for Westinghouse Pressurized Water Reactors (NUREG-0452) for the control room emergency filtration.

Items described in this response are listed, as appropriate, in Table 17A-1 under Items 23, 36, 37, 38 and 41.

COMANCHE PEAK NUCLEAR POWER PLANT
FINAL SAFETY ANALYSIS REPORT (FSAR)

PREFACE

The Effective Page Listing indicates which pages, tables and figures are current in the CPNPP/ FSAR. All pages/sheets within a FSAR section or FSAR table have an amendment number in the footer. Each sheet of a FSAR figure, which is not an “M” or “E” series station drawing, will be identified with an individual amendment number, date or “original”. The effective revision number and amendment number for “M” or “E” series figures in the FSAR are identified in Table 3.2-3. FSAR Amendment numbers are not shown on “E” and “M” series figures themselves, and the current versions are identified by revision number only.

In the EPL, a table number is preceded by a "T" (e.g., "T1.3-1") and a figure is preceded by an "F" (e.g., "F9.1-3"). Text portions of the FSAR are identified by section number only (e.g., “Section 5.4”).

For text and tables, the amendment number and/or date is generally located at the bottom of the sheet, near the margin. For non “E” and “M” series figures, the amendment number and/or date is generally located in the vicinity of the title block for that figure.

BELOW IS A LISTING OF ALL FSAR AMENDMENT NUMBERS AND DATES:

ORIGINAL	Submitted February 27, 1978
MARCH 21, 1978	Update
AMENDMENT 1	June 15, 1978
AMENDMENT 2	July 27, 1978
AMENDMENT 3	November 30, 1978
AMENDMENT 4	January 31, 1979
AMENDMENT 5	March 30, 1979
AMENDMENT 6	May 31, 1979
AMENDMENT 7	July 31, 1979
AMENDMENT 8	November 30, 1979
AMENDMENT 9	January 31, 1980
AMENDMENT 10	March 31, 1980
AMENDMENT 11	July 31, 1980
AMENDMENT 12	October 8, 1980 (Also Original Submittal of Emergency Plan)
AMENDMENT 13	December 15, 1980
AMENDMENT 14	January 30, 1981
AMENDMENT 15	February 20, 1981
AMENDMENT 16	March 31, 1981 (Also Revision 1 to Emergency Plan)
AMENDMENT 17	April 7, 1981
AMENDMENT 18	April 21, 1981
AMENDMENT 19	April 30, 1981
AMENDMENT 20	May 7, 1981
AMENDMENT 21	May 21, 1981

AMENDMENT 22	June 12, 1981
AMENDMENT 23	July 1, 1981
AMENDMENT 24	July 17, 1981
AMENDMENT 25	August 7, 1981
AMENDMENT 26	August 25, 1981
AMENDMENT 27	October 2, 1981
AMENDMENT 28	October 26, 1981 (Revision 2 to Emergency Plan)
AMENDMENT 29	December 21, 1981
AMENDMENT 30	February 1, 1982
AMENDMENT 31	April 23, 1982
AMENDMENT 32	May 21, 1982 (Also Revision 3 to Emergency Plan)
AMENDMENT 33	July 23, 1982
AMENDMENT 34	August 20, 1982 (Revision 4 to Emergency Plan)
AMENDMENT 35	October 12, 1982 (Revision 5 to Emergency Plan)
AMENDMENT 36	December 10, 1982
AMENDMENT 37	December 17, 1982
AMENDMENT 38	February 14, 1983
AMENDMENT 39	March 8, 1983 (Also Revision 6 to Emergency Plan)
AMENDMENT 40	May 10, 1983
AMENDMENT 41	July 11, 1983
AMENDMENT 42	September 12, 1983
AMENDMENT 43	August 29, 1983 (Revision 7 to Emergency Plan)
AMENDMENT 44	October 10, 1983
AMENDMENT 45	November 7, 1983
AMENDMENT 46	February 10, 1984
AMENDMENT 47	April 16, 1984
AMENDMENT 48	April 30, 1984 (Revision 8 to Emergency Plan)
AMENDMENT 49	June 5, 1984
AMENDMENT 50	July 13, 1984
AMENDMENT 51	August 6, 1984
AMENDMENT 52	August 27, 1984
AMENDMENT 53	November 5, 1984
AMENDMENT 54	January 21, 1984
AMENDMENT 55	July 19, 1985
AMENDMENT 56	October 15, 1985
AMENDMENT 57	December 20, 1985
AMENDMENT 58	June 30, 1986
AMENDMENT 59	June 1, 1986
AMENDMENT 60	November 3, 1986
AMENDMENT 61	December 19, 1986
AMENDMENT 62	March 31, 1987
AMENDMENT 63	June 15, 1987
AMENDMENT 64	July 31, 1987
AMENDMENT 65	November 20, 1987
AMENDMENT 66	January 15, 1988
AMENDMENT 67	February 5, 1988
AMENDMENT 68	February 15, 1988
AMENDMENT 69	March 14, 1988
AMENDMENT 70	April 22, 1988
AMENDMENT 71	May 27, 1988

AMENDMENT 72	July 1, 1988
AMENDMENT 73	August 5, 1988
AMENDMENT 74	October 14, 1988
AMENDMENT 75	November 18, 1988
AMENDMENT 76	May 1, 1989
AMENDMENT 77	September 8, 1989
AMENDMENT 78	January 15, 1990
AMENDMENT 79	July 31, 1990
AMENDMENT 80	November 30, 1990
AMENDMENT 81	March 15, 1991
AMENDMENT 82	July 31, 1991
AMENDMENT 83	December 13, 1991
AMENDMENT 84	February 28, 1992
AMENDMENT 85	May 29, 1992
AMENDMENT 86	August 31, 1992
AMENDMENT 87	December 18, 1992
AMENDMENT 88	April 16, 1993
AMENDMENT 89	August 20, 1993
AMENDMENT 90	December 17, 1993
AMENDMENT 91	April 15, 1994
AMENDMENT 92	August 31, 1994
AMENDMENT 93	February 1, 1995 (Updated FSAR Amendment)
AMENDMENT 94	August 1, 1996
AMENDMENT 94 ERRATA	August 1, 1996
AMENDMENT 95	February 2, 1998
AMENDMENT 96	August 2, 1999
AMENDMENT 97	February 1, 2001 (Issuance of electronc FSAR format)
AMENDMENT 98	August 1, 2002
AMENDMENT 99	February 2, 2004
AMENDMENT 100	August 1, 2005
AMENDMENT 101	February 1, 2007
AMENDMENT 102	August 1, 2008
AMENDMENT 103	February 11, 2010
AMENDMENT 104	August 1, 2011
AMENDMENT 105	February 4, 2013
AMENDMENT 106	July 31, 2014
AMENDMENT 107	February 1, 2016

FSAR Chapter	Amendment No.
Chapter 1 TOC	104
Chapter 1 LOT	104
Chapter 1 LOF	104
Section 1.1	106
Section 1.2	107
F1.2-1	107
F1.2-2	10
F1.2-3	10
F1.2-4	10
F1.2-5	10
F1.2-6	10
F1.2-7	10
F1.2-8	102
F1.2-9	10
F1.2-10	62
F1.2-11	84
F1.2-12	10
F1.2-13	102
F1.2-14	10
F1.2-15	10
F1.2-16	10
F1.2-17	10
F1.2-18	10
F1.2-19	10
F1.2-20	10
F1.2-21	10
F1.2-22	10
F1.2-23	10
F1.2-24	10
F1.2-25	10
F1.2-26	10
F1.2-27	10
F1.2-28	10
F1.2-29	10
F1.2-30	10
F1.2-31	10
F1.2-32	10
F1.2-33	100
F1.2-34	10
F1.2-35	10
F1.2-36	10
F1.2-37	10
F1.2-38	93
F1.2-39	10
F1.2-40	10
F1.2-41	10
F1.2-42	10
F1.2-43	10
F1.2-44	10
F1.2-45	10
F1.2-46	10
Section 1.3	104
T1.3-1	Deleted
T1.3-2	104
Section 1.4	104
T1.4-1	104
Section 1.5	104
T1.5-1	104
T1.5-2	104
Section 1.6	104
T1.6-1	105
T1.6-2	104
Section 1.7	104
T1.7-1	104
T1.7-2	104
Appendix 1A(N)	104
T1A(N)-1	104
F1A(N)-1	Original
F1A(N)-2	Original
Appendix 1A(B)	107
Chapter 2 TOC	104
Section 2.1	104
T2.1-1	104
T2.1-2	104
T2.1-3	104
T2.1-4	104
T2.1-5	104
T2.1-6	104
T2.1-7	104

(continued)

Page	Amendment No.
T2.1-8	104
T2.1-9	104
F2.1-1	Original
F2.1-2	89
F2.1-2A	87
F2.1-2B	87
F2.1-2C	89
F2.1-3	Original
F2.1-4	Original
F2.1-5	53
F2.1-6	Original
F2.1-7	Original
Section 2.2	104
T2.2-1	104
T2.2-2	104
F2.2-1	82
Section 2.3	104
T2.3-1	104
T2.3-2	104
T2.3-2A	104
T2.3-3	104
T2.3-4	104
T2.3-5	104
T2.3-6	104
T2.3-7	Deleted
T2.3-7A	104
T2.3-7B	104
T2.3-7C	104
T2.3-8	Deleted
T2.3-9	Deleted
T2.3-10	Deleted
T2.3-11	Deleted
T2.3-12	104
T2.3-13	104
T2.3-14	104
T2.3-15	104
T2.3-16	104
T2.3-17	104
T2.3-18	104
T2.3-19	104
T2.3-20	104
T2.3-21	104
T2.3-22	104

Page	Amendment No.
T2.3-24	104
T2.3-25	104
T2.3-26	104
T2.3-28	104
T2.3-29	Deleted
T2.3-30	Deleted
T2.3-31	Deleted
T2.3-32	Deleted
T2.3-33	104
T2.3-34	104
T2.3-34A	104
T2.3-34B	104
T2.3-35	104
T2.3-36	Deleted
T2.3-37	104
T2.3-38	104
F2.3-1	Original
F2.3-2	Original
F2.3-3	Original
F2.3-4	Original
F2.3-5	Original
F2.3-6	Original
F2.3-7	Original
F2.3-8	Original
F2.3-9	Original
F2.3-10	Original
F2.3-11	Original
F2.3-12	87
F2.3-13	94
F2.3-14	91
F2.3-15	91
F2.3-16	91
F2.3-17	91
F2.3-18	Original
F2.3-19	Original
Section 2.4	106
T2.4-1	104
T2.4-2	104
T2.4-3	104
T2.4-4	104
T2.4-5	104
T2.4-6	104
T2.4-7	104
T2.4-8	104

(continued)

Page	Amendment No.
T2.4-9	104
T2.4-10	104
T2.4-11	104
T2.4-12	104
T2.4-13	104
T2.4-14	104
T2.4-15	104
T2.4-16	104
T2.4-17	104
T2.4-18	104
T2.4-19	104
T2.4-20	104
T2.4-20A	104
T2.4-21	104
T2.4-22	104
T2.4-23	104
T2.4-24	104
T2.4-25	104
T2.4-26	104
T2.4-27	104
T2.4-28	104
T2.4-29	104
F2.4-1	Original
F2.4-2	7
F2.4-3	Original
F2.4-4	87
F2.4-5	Original
F2.4-6	Original
F2.4-7	Original
F2.4-8	Original
F2.4-9	78
F2.4-10	Original
F2.4-11	Original
F2.4-12	Original
F2.4-13	Original
F2.4-14	Original
F2.4-15	87
F2.4-16	Original
F2.4-17	Original
F2.4-18	Original
F2.4-19	Original
F2.4-20	Original
F2.4-21	Original
F2.4-22	10
F2.4-23	Original

Page	Amendment No.
F2.4-24	Original
F2.4-25	Original
F2.4-26	87
F2.4-27	87
F2.4-28	87
F2.4-29	87
F2.4-30	Original
F2.4-31	Original
F2.4-32	Original
F2.4-33	Original
F2.4-34	Original
F2.4-35	Original
F2.4-36	Original
F2.4-37	Original
F2.4-38	Original
F2.4-39	Original
F2.4-40	1
F2.4-41	1
F2.4-42	July 31, 1979
Section 2.5	104
T2.5.1-1	104
T2.5.1-2	104
T2.5.1-3	104
T2.5.1-4	104
T2.5.1-5	104
T2.5.1-6	104
T2.5.1-7	104
T2.5.1-8	104
T2.5.1-9	104
T2.5.1-10	104
T2.5.1-11	104
T2.5.2-1	104
T2.5.2-2	104
T2.5.2-3	104
T2.5.4-1	104
T2.5.4-2	104
T2.5.4-3	104
T2.5.4-4	104
T2.5.4-5	104
T2.5.4-5A	104
T2.5.4-5B	104
T2.5.4-5C	104
T2.5.4-5D	104
T2.5.4-5E	104

(continued)

Page	Amendment No.
T2.5.4-5F	104
T2.5.4-5G	104
T2.5.4-5H	104
T2.5.4-6	104
T2.5.4-6A	104
T2.5.4-7	104
T2.5.4-8	104
T2.5.4-9	104
T2.5.4-10	104
T2.5.4-11	104
T2.5.4-12	104
T2.5.4-13	104
T2.5.4-13A	104
T2.5.6-1	104
T2.5.6-2	104
T2.5.6-3	104
T2.5.6-4	104
T2.5.6-5	104
T2.5.6-6	104
T2.5.6-7	104
T2.5.6-8	104
F2.5.1-1	Original
F2.5.1-2	67
F2.5.1-3	7
F2.5.1-4	2
F2.5.1-4a	7
F2.5.1-5	7
F2.5.1-6	7
F2.5.1-7	67
F2.5.1-8	Original
F2.5.1-9	Original
F2.5.1-10	2
F2.5.1-11	67
F2.5.1-12	87
F2.5.1-13	67
F2.5.1-14	67
F2.5.1-15	87
F2.5.1-16	67
F2.5.1-17	16
F2.5.1-18	Original
F2.5.1-19	Original
F2.5.1-20	Original
F2.5.1-21	Original
F2.5.1-22	4
F2.5.1-23	Original
F2.5.1-23a	July 27, 1978
F2.5.1-24	2
F2.5.1-24a	16
F2.5.1-25	2
F2.5.1-26	2
F2.5.1-27	2
F2.5.1-28	2
F2.5.1-29	2
F2.5.1-30	2
F2.5.1-31	2
F2.5.1-32	67
F2.5.1-33	4
F2.5.1-34	4
F2.5.1-35	4
F2.5.2-1	Original
F2.5.2-2	Original
F2.5.2-3	Original
F2.5.2-4	Original
F2.5.2-5	Original
F2.5.2-6	Original
F2.5.2-7	Original
F2.5.2-8	Original
F2.5.2-9	Original
F2.5.2-10	Original
F2.5.2-11	Original
F2.5.2-12	Original
F2.5.2-13	Original
F2.5.2-14	Original
F2.5.4-1 (Sh. 1)	Original
F2.5.4-1 (Sh. 2)	Original
F2.5.4-1 (Sh. 3)	Original
F2.5.4-1 (Sh. 4)	Original
F2.5.4-1 (Sh. 5)	Original
F2.5.4-1 (Sh. 6)	Original
F2.5.4-1 (Sh. 7)	Original
F2.5.4-1 (Sh. 8)	Original
F2.5.4-1 (Sh. 9)	Original
F2.5.4-1 (Sh. 10)	Original
F2.5.4-1 (Sh. 11)	Original
F2.5.4-1 (Sh. 12)	Original
F2.5.4-1 (Sh. 13)	Original
F2.5.4-1 (Sh. 14)	Original
F2.5.4-1 (Sh. 15)	Original
F2.5.4-1 (Sh. 16)	Original
F2.5.4-1 (Sh. 17)	Original

(continued)

Page	Amendment No.	Page	Amendment No.
F2.5.4-1 (Sh. 18)	Original	F2.5.4-34	Original
F2.5.4-1 (Sh. 19)	Original	F2.5.4-35	Original
F2.5.4-1 (Sh. 20)	Original	F2.5.4-36	Original
F2.5.4-1 (Sh. 21)	Original	F2.5.4-37 (Sh. 1)	Original
F2.5.4-1 (Sh. 22)	Original	F2.5.4-37 (Sh. 2)	Original
F2.5.4-1 (Sh. 23)	Original	F2.5.4-37 (Sh. 3)	Original
F2.5.4-1 (Sh. 24)	Original	F2.5.4-37 (Sh. 4)	Original
F2.5.4-1 (Sh. 25)	Original	F2.5.4-37 (Sh. 5)	Original
F2.5.4-2	Original	F2.5.4-37 (Sh. 6)	Original
F2.5.4-3	Original	F2.5.4-37 (Sh. 7)	Original
F2.5.4-4	Original	F2.5.4-37 (Sh. 8)	Original
F2.5.4-5	Original	F2.5.4-37 (Sh. 9)	Original
F2.5.4-6	Original	F2.5.4-37 (Sh. 10)	Original
F2.5.4-7	Original	F2.5.4-37 (Sh. 11)	Original
F2.5.4-8	Original	F2.5.4-37 (Sh. 12)	Original
F2.5.4-9	Original	F2.5.4-37 (Sh. 13)	Original
F2.5.4-10	Original	F2.5.4-37 (Sh. 14)	Original
F2.5.4-11	Original	F2.5.4-37 (Sh. 15)	Original
F2.5.4-12	Original	F2.5.4-37 (Sh. 16)	Original
F2.5.4-13	Original	F2.5.4-37 (Sh. 17)	Original
F2.5.4-14	67	F2.5.4-37 (Sh. 18)	Original
F2.5.4-14A	67	F2.5.4-37 (Sh. 19)	Original
F2.5.4-15	Original	F2.5.4-37 (Sh. 20)	Original
F2.5.4-16	Original	F2.5.4-37 (Sh. 21)	Original
F2.5.4-17	Original	F2.5.4-37A	22
F2.5.4-18	Original	F2.5.4-38	2
F2.5.4-19	Original	F2.5.4-38A	2
F2.5.4-20	Original	F2.5.4-38B	2
F2.5.4-21	67	F2.5.4-38C	2
F2.5.4-22	Original	F2.5.4-38D	2
F2.5.4-23	Original	F2.5.4-39 (Sh. 1)	67
F2.5.4-24	Original	F2.5.4-39 (Sh. 2)	67
F2.5.4-25	Original	F2.5.4-39 (Sh. 3)	67
F2.5.4-26	Original	F2.5.4-39 (Sh. 4)	67
F2.5.4-27	67	F2.5.4-40	67
F2.5.4-28	87	7F2.5.4-41	4
F2.5.4-28A	March 31, 1980	F2.5.4-42	67
F2.5.4-29	87	F2.5.4-43	67
F2.5.4-30	Original	F2.5.4-44	67
F2.5.4-30A (Sheet 1)	October 14, 1988	F2.5.4-45	67
F2.5.4-30A (Sheet 2)	October 14, 1988	F2.5.4-46	67
F2.5.4-31	67	F2.5.4-47	67
F2.5.4-32	95	F2.5.4-48	67
F2.5.4-33A	67	F2.5.4-49	July 31, 1980
F2.5.4-33B	67	F2.5.4-50	July 31, 1980

(continued)

Page	Amendment No.	Page	Amendment No.
F2.5.4-51	July 31, 1980	F2.5.5-37	Original
F2.5.4-52	July 31, 1980	F2.5.5-38	Original
F2.5.4-53	July 31, 1980	F2.5.5-39	Original
F2.5.4-54	October 14, 1988	F2.5.5-40	Original
F2.5.4-55	87	F2.5.5-41	Original
F2.5.4-56	67	F2.5.5-42	Original
F2.5.4-57	75	F2.5.5-43	Original
F2.5.5-1	4	F2.5.5-44	Original
F2.5.5-2	4	F2.5.5-45	Original
F2.5.5-3	4	F2.5.5-46	Original
F2.5.5-4	4	F2.5.5-47	Original
F2.5.5-5	67	F2.5.5-48	Original
F2.5.5-6	Original	F2.5.5-49	Original
F2.5.5-7	Original	F2.5.5-50	January 31, 1979
F2.5.5-8	Original	F2.5.5-51	Original
F2.5.5-9	Original	F2.5.5-52	Original
F2.5.5-10	Original	F2.5.5-53	Original
F2.5.5-11	Original	F2.5.5-54	Original
F2.5.5-12	Original	F2.5.5-55	Original
F2.5.5-13	Original	F2.5.5-56	Original
F2.5.5-14	Original	F2.5.5-57	Original
F2.5.5-15	Original	F2.5.5-58	Original
F2.5.5-16 (Sh. 1)	67	F2.5.5-59	Original
F2.5.5-16 (Sh. 2)	67	F2.5.5-59a	Original
F2.5.5-17	Original	F2.5.5-59b	Original
F2.5.5-18 (Sh. 1)	Original	F2.5.5-59c	Original
F2.5.5-18 (Sh. 2)	Original	F2.5.5-60	Original
F2.5.5-19	Original	F2.5.5-61	Original
F2.5.5-20	Original	F2.5.5-62	Original
F2.5.5-21	Original	F2.5.5-63	Original
F2.5.5-22	Original	F2.5.5-64	Original
F2.5.5-23	Original	F2.5.5-65	Original
F2.5.5-24	Original	F2.5.5-66	Original
F2.5.5-25	Original	F2.5.5-67	Original
F2.5.5-26	Original	F2.5.5-68	Original
F2.5.5-27	Original	F2.5.5-69	Original
F2.5.5-28	Original	F2.5.5-70	Original
F2.5.5-29	Original	F2.5.5-71	Original
F2.5.5-30	Original	F2.5.5-72	Original
F2.5.5-31	Original	F2.5.5-73	Original
F2.5.5-32	Original	F2.5.5-74	Original
F2.5.5-33	Original	F2.5.5-75	Original
F2.5.5-34	Original	F2.5.5-76 (Sh. 1)	Original
F2.5.5-35	Original	F2.5.5-76 (Sh. 2)	Original
F2.5.5-36	67	F2.5.5-76 (Sh. 3)	Original

(continued)

Page	Amendment No.
F2.5.5-77	Original
F2.5.5-78 (Sh. 1)	78
F2.5.5-78 (Sh. 2)	78
F2.5.5-79 (Sh. 1)	78
F2.5.5-79 (Sh. 2)	78
F2.5.5-80 (Sh. 1)	78
F2.5.5-80 (Sh. 2)	78
F2.5.5-81 (Sh. 1)	78
F2.5.5-81 (Sh. 2)	78
F2.5.5-82 (Sh. 1)	78
F2.5.5-82 (Sh. 2)	78
F2.5.5-83 (Sh. 1)	78
F2.5.5-83 (Sh. 2)	78
F2.5.5-84 (Sh. 1)	78
F2.5.5-84 (Sh. 2)	78
F2.5.5-85 (Sh. 1)	78
F2.5.5-85 (Sh. 2)	78
F2.5.5-86 (Sh. 1)	78
F2.5.5-86 (Sh. 2)	78
F2.5.5-87 (Sh. 1)	78
F2.5.5-87 (Sh. 2)	78
F2.5.5-88 (Sh. 1)	78
F2.5.5-88 (Sh. 2)	78
F2.5.5-89 (Sh. 1)	78
F2.5.5-89 (Sh. 2)	78
F2.5.5-90 (Sh. 1)	78
F2.5.5-90 (Sh. 2)	78
F2.5.5-91 (Sh. 1)	78
F2.5.5-91 (Sh. 2)	78
F2.5.5-92	78
F2.5.5-93	78
F2.5.5-94	78
F2.5.6-1	Original
F2.5.6-2	Original
F2.5.6-3	Original
F2.5.6-4A	Original
F2.5.6-4B	Original
F2.5.6-4C	Original
F2.5.6-4D	Original
F2.5.6-4E	Original
F2.5.6-4F	Original
F2.5.6-4G	Original
F2.5.6-5A	Original
F2.5.6-5B	Original
F2.5.6-5C	Original
F2.5.6-5D	Original
F2.5.6-5E	Original
F2.5.6-5F	Original
F2.5.6-5G	Original
F2.5.6-5H	Original
F2.5.6-5I	Original
F2.5.6-5J	Original
F2.5.6-7	67
F2.5.6-8	Original
F2.5.6-9	Original
F2.5.6-10	67
F2.5.6-11	67
F2.5.6-12	4
F2.5.6-13	Original
F2.5.6-14	Original
F2.5.6-15	Original
F2.5.6-16	Original
F2.5.6-17	Original
F2.5.6-18	Original
F2.5.6-19	Original
F2.5.6-20	Original
F2.5.6-21	Original
F2.5.6-22	Original
F2.5.6-23	Original
F2.5.6-24	2
F2.5.6-25	Original
F2.5.6-26	Original
F2.5.6-27	Original
F2.5.6-28	Original
F2.5.6-29	Original
F2.5.6-30	Original
F2.5.6-31	Original
F2.5.6-32	Original
F2.5.6-33	Original
F2.5.6-34	Original
F2.5.6-35	Original
F2.5.6-36	Original
F2.5.6-37	Original
F2.5.6-38	Original
F2.5.6-39	Original
F2.5.6-40	Original
F2.5.6-41	Original
F2.5.6-42	Original
F2.5.6-43	Original
F2.5.6-44	Original

(continued)

Page	Amendment No.
F2.5.6-45	Original
F2.5.6-46	2
F2.5.6-47	67
F2.5.6-48	67
F2.5.6-49	Original
F2.5.6-50	Original
F2.5.6-51	Original
F2.5.6-52	Original
F2.5.6-53	Original
F2.5.6-54	Original
F2.5.6-55	Original
F2.5.6-56	103
Section 2.5A	104
T2.5A-1	104
T2.5A-2	104
T2.5A-3	104
T2.5A-4	104
F2.5A-1	June 15, 1978
F2.5A-2	June 15, 1978
F2.5A-3	June 15, 1978
F2.5A-4	June 15, 1978
F2.5A-5	June 15, 1978
F2.5A-6	June 15, 1978
F2.5A-7	June 15, 1978
F2.5A-8	June 15, 1978
F2.5A-9	June 15, 1978
F2.5A-10	June 15, 1978
F2.5A-11 (Sh. 1)	June 15, 1978
F2.5A-11 (Sh. 2)	June 15, 1978
F2.5A-11 (Sh. 3)	June 15, 1978
F2.5A-11 (Sh. 4)	June 15, 1978
F2.5A-11 (Sh. 5)	June 15, 1978
F2.5A-11 (Sh. 6)	June 15, 1978
F2.5A-11 (Sh. 7)	June 15, 1978
F2.5A-11 (Sh. 8)	June 15, 1978
F2.5A-11 (Sh. 9)	June 15, 1978
F2.5A-11 (Sh. 10)	June 15, 1978
F2.5A-11 (Sh. 11)	June 15, 1978
F2.5A-11 (Sh. 12)	June 15, 1978
F2.5A-11 (Sh. 13)	June 15, 1978
F2.5A-11 (Sh. 14)	June 15, 1978
F2.5A-11 (Sh. 15)	June 15, 1978
F2.5A-12	June 15, 1978
F2.5A-13	June 15, 1978

Page	Amendment No.
F2.5A-14	June 15, 1978
F2.5A-15	June 15, 1978
F2.5A-16	June 15, 1978
F2.5A-17	4
F2.5A-18	June 15, 1978
F2.5A-19	4
F2.5A-20	4
F2.5A-21	4
Section 2.5B	104
T2.5B-1	104
T2.5B-2	104
T2.5B-3	104
T2.5B-4	104
T2.5B-5	104
F2.5B-1	June 15, 1978
F2.5B-2	June 15, 1978
F2.5B-3	June 15, 1978
F2.5B-4	June 15, 1978
F2.5B-5	June 15, 1978
F2.5B-6	June 15, 1978
F2.5B-7 (Sh. 1)	June 15, 1978
F2.5B-7 (Sh. 2)	June 15, 1978
F2.5B-7 (Sh. 3)	June 15, 1978
F2.5B-7 (Sh. 4)	June 15, 1978
F2.5B-7 (Sh. 5)	June 15, 1978
F2.5B-7 (Sh. 6)	June 15, 1978
F2.5B-7 (Sh. 7)	June 15, 1978
F2.5B-7 (Sh. 8)	June 15, 1978
F2.5B-7 (Sh. 9)	June 15, 1978
F2.5B-7 (Sh. 10)	June 15, 1978
F2.5B-7 (Sh. 11)	June 15, 1978
F2.5B-7 (Sh. 12)	June 15, 1978
F2.5B-7 (Sh. 13)	June 15, 1978
F2.5B-7 (Sh. 14)	June 15, 1978
F2.5B-7 (Sh. 15)	June 15, 1978
F2.5B-7 (Sh. 16)	June 15, 1978
F2.5B-7 (Sh. 17)	June 15, 1978
F2.5B-7 (Sh. 18)	June 15, 1978
F2.5B-7 (Sh. 19)	June 15, 1978
F2.5B-7 (Sh. 20)	June 15, 1978
F2.5B-7 (Sh. 21)	June 15, 1978
F2.5B-7 (Sh. 22)	June 15, 1978
F2.5B-7 (Sh. 23)	June 15, 1978
F2.5B-7 (Sh. 24)	June 15, 1978

(continued)

Page	Amendment No.
F2.5B-7 (Sh. 25)	June 15, 1978
F2.5B-7 (Sh. 26)	June 15, 1978
F2.5B-7 (Sh. 27)	June 15, 1978
F2.5B-7 (Sh. 28)	June 15, 1978
F2.5B-7 (Sh. 29)	June 15, 1978
F2.5B-7 (Sh. 30)	June 15, 1978
F2.5B-7 (Sh. 31)	June 15, 1978
F2.5B-7 (Sh. 32)	June 15, 1978
F2.5B-7 (Sh. 33)	June 15, 1978
F2.5B-8	June 15, 1978
F2.5B-9	June 15, 1978
F2.5B-10	June 15, 1978
Chapter 3 TOC	104
Chapter 3 LOT	104
Chapter 3 LOF	104
Section 3.1	104
Section 3.2	104
T3.2-1	104
T3.2-2	104
T3.2-3	107
T3.2-4	107
F3.2-1	94
Section 3.3	104
T3.3-1	Deleted
F3.3-1	42
F3.3-2	42
F3.3-3	42
Section 3.4	105
T3.4-1	104
Section 3.5	106
T3.5-1	104
T3.5-2	Deleted
T3.5-2A	104
T3.5-2B	104
T3.5-3	104
T3.5-4	104
T3.5-5	104
T3.5-6	104
T3.5-7	104

Page	Amendment No.
T3.5-8	104
T3.5-9	Deleted
T3.5-10	Deleted
F3.5-1	102
F3.5-2	102
F3.5-3	24
Section 3.5A	Deleted
Section 3.5B	Deleted
Section 3.6N	Deleted
Section 3.6B	106
T3.6B-1	104
T3.6B-2	104
T3.6B-3	Deleted
T3.6B-4	Deleted
T3.6B-4A	104
T3.6B-4B	104
T3.6B-5	104
T3.6B-6	104
T3.6B-7	Deleted
T3.6B-8	Deleted
F3.6B-1	11
F3.6B-2	11
F3.6B-3	11
F3.6B-4	11
F3.6B-5	73
F3.6B-6	July 31, 1980
F3.6B-7	July 31, 1980
F3.6B-8	July 31, 1980
F3.6B-9	76
F3.6B-10	July 31, 1980
F3.6B-11	102
F3.6B-12	102
F3.6B-13	102
F3.6B-14	102
F3.6B-15	78
F3.6B-16	78
F3.6B-17	78
F3.6B-18	78
F3.6B-19	102
F3.6B-20	102
F3.6B-21	102

(continued)

Page	Amendment No.	Page	Amendment No.
F3.6B-22	102	F3.6B-53	80
F3.6B-23-1	78	F3.6B-54	78
F3.6B-23-2	78	F3.6B-55	102
F3.6B-23-3	76	F3.6B-56	76
F3.6B-24-1	78	F3.6B-57-1	76
F3.6B-24-2	76	F3.6B-57-2	76
F3.6B-24-3	76	F3.6B-58-1	76
F3.6B-24-4	76	F3.6B-58-2	76
F3.6B-25-1	76	F3.6B-58-3	78
F3.6B-25-2	76	F3.6B-59-1	78
F3.6B-25-3	76	F3.6B-59-2	76
F3.6B-25-4	76	F3.6B-60	76
F3.6B-26	102	F3.6B-61-1	76
F3.6B-27	102	F3.6B-61-2	76
F3.6B-28	102	F3.6B-62	76
F3.6B-29	102	F3.6B-63	76
F3.6B-30	102	F3.6B-64-1	101
F3.6B-31-1	91	F3.6B-64-2	101
F3.6B-31-2	96	F3.6B-65	78
F3.6B-32	76	F3.6B-66-1	76
F3.6B-33	41	F3.6B-66-2	76
F3.6B-34	102	F3.6B-67-1	102
F3.6B-35	102	F3.6B-67-2	76
F3.6B-36	102	F3.6B-67-3	76
F3.6B-37	102	F3.6B-68-1	80
F3.6B-38	102	F3.6B-68-2	80
F3.6B-39-1	102	F3.6B-69	76
F3.6B-39-2	76	F3.6B-70	76
F3.6B-40	102	F3.6B-71	76
F3.6B-41	102	F3.6B-72	76
F3.6B-42	76	F3.6B-73-1	76
F3.6B-43	80	F3.6B-73-2	76
F3.6B-44	78	F3.6B-74	80
F3.6B-45	76	F3.6B-75	76
F3.6B-46	26	F3.6B-76	26
F3.6B-47	26	F3.6B-77	26
F3.6B-48-1	80	F3.6B-78	76
F3.6B-48-2	80	F3.6B-79	76
F3.6B-49	80	F3.6B-80	78
F3.6B-50-1	76	F3.6B-81	76
F3.6B-50-2	76	F3.6B-82-1	102
F3.6B-51-1	76	F3.6B-82-2	76
F3.6B-51-2	76	F3.6B-82-3	76
F3.6B-52-1	80	F3.6B-83	76
F3.6B-52-2	80	F3.6B-84	78

(continued)

Page	Amendment No.	Page	Amendment No.
F3.6B-85	76	F3.6B-127	86
F3.6B-86	76	F3.6B-128	86
F3.6B-87-1	76	F3.6B-129	96
F3.6B-87-2	76	F3.6B-130	86
F3.6B-88-1	76	F3.6B-132	86
F3.6B-88-2	76	F3.6B-133	86
F3.6B-89	76	F3.6B-134	86
F3.6B-90	76	F3.6B-135	86
F3.6B-91	76	F3.6B-136	86
F3.6B-92	76	F3.6B-137	86
F3.6B-93-1	87	F3.6B-138	86
F3.6B-93-2	87	F3.6B-139	86
F3.6B-94	87	F3.6B-140	86
F3.6B-95	87	F3.6B-141	88
F3.6B-96A	71	F3.6B-142	88
F3.6B-96B	71	F3.6B-143	86
F3.6B-96C	71	F3.6B-144	86
F3.6B-96D	76	F3.6B-147	86
F3.6B-96E	76	F3.6B-148	86
F3.6B-100	86	F3.6B-149	86
F3.6B-101	86	F3.6B-150	86
F3.6B-102	86	F3.6B-151	86
F3.6B-103	86	F3.6B-152	86
F3.6B-104	86	F3.6B-153	86
F3.6B-105	86	F3.6B-154	86
F3.6B-106	86	F3.6B-155	86
F3.6B-107	86	F3.6B-156	86
F3.6B-108	86	F3.6B-157	86
F3.6B-109	86	F3.6B-158	86
F3.6B-110	86	F3.6B-159	86
F3.6B-111	86	F3.6B-160	86
F3.6B-112	86	F3.6B-161	86
F3.6B-113	86	F3.6B-162	86
F3.6B-114	87	F3.6B-163	86
F3.6B-115	87	F3.6B-164	86
F3.6B-116	86	F3.6B-165	86
F3.6B-117	86	F3.6B-166	86
F3.6B-118	86	F3.6B-167	86
F3.6B-119	86	F3.6B-168	86
F3.6B-120	87	F3.6B-169	86
F3.6B-121	86	F3.6B-172	86
F3.6B-122	86	F3.6B-173	86
F3.6B-124	96	F3.6B-175	86
F3.6B-125	96	F3.6B-176	86
F3.6B-126	96	F3.6B-177	86

(continued)

Page	Amendment No.
F3.6B-178	86
F3.6B-179	86
F3.6B-180	86
F3.6B-181	86
F3.6B-182	86
F3.6B-183	86
F3.6B-184	88
F3.6B-185	86
F3.6B-188	87
F3.6B-189	87
F3.6B-192	86
F3.6B-193	86
F3.6B-194	86
F3.6B-195	86
F3.6B-196	86
F3.6B-197	86
F3.6B-198	86
F3.6B-199	86
F3.6B-200	86
F3.6B-201	86
F3.6B-202	86
F3.6B-203	86
F3.6B-204	86
F3.6B-205	86
F3.6B-206	86
F3.6B-207 (Sh. 1)	93
F3.6B-207 (Sh. 2)	93
F3.6B-207 (Sh. 3)	93
F3.6B-207 (Sh. 4)	93
F3.6B-208 (Sh. 1)	91
F3.6B-208 (Sh. 2)	102
F3.6B-208 (Sh. 3)	102
Section 3.7N	104
T3.7N-1	104
F3.7N-1	Original
F3.7N-2	Original
F3.7N-3	Original
F3.7N-4	20
F3.7N-5	20
Section 3.7B	104
T3.7B-1	104
T3.7B-2	104
T3.7B-3	104
T3.7B-3A	104
T3.7B-4	104
T3.7B-5	104
T3.7B-6	104
T3.7B-7	104
T3.7B-8	104
T3.7B-9	104
T3.7B-10	104
T3.7B-11	104
T3.7B-12	104
T3.7B-12A	104
T3.7B-13	104
T3.7B-14	104
T3.7B-15	104
T3.7B-16	104
T3.7B-17	104
T3.7B-17A	104
T3.7B-18	104
T3.7B-19	104
T3.7B-20	104
T3.7B-21	104
T3.7B-22	104
T3.7B-23	104
T3.7B-24	104
T3.7B-25	104
T3.7B-26	104
T3.7B-27	104
T3.7B-28	104
T3.7B-29	104
T3.7B-30	104
T3.7B-31	104
T3.7B-32	104
T3.7B-33	104
T3.7B-34	104
T3.7B-35	104
T3.7B-36	104
T3.7B-37	104
T3.7B-38	104
T3.7B-39	104
T3.7B-40	104
T3.7B-41	104
T3.7B-42	104
T3.7B-43	104
T3.7B-44	104
T3.7B-45	104

(continued)

Page	Amendment No.
T3.7B-46	104
T3.7B-47	104
T3.7B-48	104
T3.7B-49	104
T3.7B-50	104
T3.7B-51	104
T3.7B-52	104
F3.7B-1	Original
F3.7B-2	68
F3.7B-3	68
F3.7B-3A	98
F3.7B-3B	98
F3.7B-4	68
F3.7B-4A	98
F3.7B-4B	98
F3.7B-5	68
F3.7B-5A	98
F3.7B-5B	98
F3.7B-6	68
F3.7B-7	Original
F3.7B-8	68
F3.7B-9	68
F3.7B-9A	98
F3.7B-10	68
F3.7B-10A	98
F3.7B-11	68
F3.7B-11A	98
F3.7B-12	68
F3.7B-13	68
F3.7B-14	68
F3.7B-14A	98
F3.7B-14B	98
F3.7B-15	68
F3.7B-16	68
F3.7B-17	68
F3.7B-18	68
F3.7B-19	68
F3.7B-19A	98
F3.7B-20	98
F3.7B-21	98
F3.7B-22	68
F3.7B-23	Original
F3.7B-24	Original
F3.7B-25	Original
F3.7B-26	Original
F3.7B-27	Original
F3.7B-28	Original
F3.7B-29	Original
F3.7B-30	Original
F3.7B-31	Original
F3.7B-32	Original
F3.7B-33	Original
F3.7B-34	Original
F3.7B-35	Original
F3.7B-36	Original
F3.7B-37	Original
F3.7B-37A	98
F3.7B-38	68
F3.7B-39	Original
F3.7B-40	Original
F3.7B-41	78
F3.7B-42	78
F3.7B-43	78
F3.7B-44	78
F3.7B-45	78
F3.7B-46	78
F3.7B-47	78
F3.7B-48	78
F3.7B-49	78
F3.7B-50	78
F3.7B-50A	78
F3.7B-50AA	98
F3.7B-50AB	98
F3.7B-50B	78
F3.7B-50C	78
F3.7B-51	68
F3.7B-52	68
F3.7B-53	68
F3.7B-54	102
F3.7B-55	4
Section 3.7B(A)	104
T3.7B(A)-1	107
F3.7B(A)-1	Original
Section 3.8	104
T3.8-1	104
F3.8-1	Original
F3.8-2	102
F3.8-3	Original

(continued)

Page	Amendment No.
F3.8-4	Original
F3.8-5	Original
F3.8-6	Original
F3.8-7	68
F3.8-8	77
F3.8-9	68
F3.8-10	Original
F3.8-11	Original
F3.8-11A	March 31, 1980
F3.8-12	Original
F3.8-13	Original
F3.8-14	Original
F3.8-15	102
F3.8-16	8
F3.8-17	Original
F3.8-18	Original
F3.8-19	93
F3.8-19a	93
F3.8-20	5
F3.8-21	5
F3.8-22	(See Table 3.2-3)
F3.8-23	(See Table 3.2-3)
F3.8-24	91
F3.8-25	91
Appendix 3.8A	104
Figure 3.8A-3-1	102
Figure 3.8A-3-2	102
Figure 3.8A-3-3	102
Figure 3.8A-3-4	102
Section 3.9N	104
T3.9N-1	104
T3.9N-1a	104
T3.9N-2	104
T3.9N-3	104
T3.9N-4	104
T3.9N-5	104
T3.9N-6	104
T3.9N-7	104
T3.9N-8	104
T3.9N-9	104
T3.9N-10	105
T3.9N-11	104
T3.9N-12	104

Page	Amendment No.
T3.9N-13	104
T3.9N-14	Deleted
T3.9N-14A	104
T3.9N-14B	104
T3.9N-15	Deleted
T3.9N-15A	104
T3.9N-15B	104
T3.9N-16	Deleted
T3.9N-16A	104
T3.9N-16B	104
T3.9N-17	104
T3.9N-18	104
T3.9N-19	Deleted
T3.9N-19A	104
T3.9N-19B	104
T3.9N-20	104
T3.9N-21	Deleted
T3.9N-21A	104
T3.9N-21B	104
T3.9N-22	104
F3.9N-1	102
F3.9N-2	Original
F3.9N-3	Original
F3.9N-4	Original
F3.9N-4a	21
F3.9N-5	Deleted
F3.9N-5A	102
F3-9N-5B	102
F3.9N-6	Deleted
F3.9N-6A	102
F3.9N-6B	102
F3.9N-7	Original
F3.9N-8	Original
F3.9N-9	Original
F3.9N-10	Original
F3.9N-11	Original
F3.9N-12	March 31, 1980
F3.9N-13	March 31, 1980
Section 3.9B	104
T3.9B-1A	104
T3.9B-1B	104
T3.9B-1C	104
T3.9B-1D	104
T3.9B-1E	104

(continued)

Page	Amendment No.
T3.9B-1F	104
T3.9B-2	104
T3.9B-3	104
T3.9B-4	104
T3.9B-5	104
T3.9B-6	Deleted
T3.9B-7	Deleted
T3.9B-8	104
T3.9B-9	104
T3.9B-10	107
T3.9B-11	104
T3.9B-12	104
F3.9B-1	Original
F3.9B-2	Original
F3.9B-3	Original
F3.9B-4	Original
F3.9B-5	Original
F3.9B-6	74
F3.9B-7	Original
F3.9B-8	Original
Section 3.10N	104
T3.10N-1	104
Section 3.10B	104
Section 3.11N	104
T3.11N-1	Deleted
T3.11N-2	Deleted
T3.11N-3	Deleted
Section 3.11B	104
T3.11B-1	Deleted
T3.11B-2	Deleted
T3.11B-3	Deleted
T3.11B-4	Deleted
T3.11B-5	Deleted
Appendix 3A TOC	104
Appendix 3A	104
T3A.3-1	105
T3A.3-2	105
Appendix 3B	104

Page	Amendment No.
T3B.1-1	104
T3B.1-2	104
T3B.1-3	104
T3B.1-4	104
T3B.1-5	104
T3B.1-6	104
T3B.1-7	104
T3B.1-8	104
T3B.1-9	104
T3B.1-10	104
T3B.3-1	104
T3B.5-1	104
T3B.5-2	104
T3B.6-1	104
T3B.7-1	104
T3B.7-2	104
T3B.8-1	104
T3B.9-1	104
T3B.9-2	104
T3B.10-1	104
T3B.10-2	104
T3B.11-1	104
T3B.11-2	104
T3B.11-3	104
T3B.11-4	104
T3B.13-1	104
T3B.13-2	104
T3B.13-3	104
T3B.13-4	104
T3B.13-5	104
T3B.13-6	104
T3B.13-7	104
T3B.13-8	104
T3B.13-9	104
T3B.13-10	104
T3B.13-11	104
T3B.14-1	104
T3B.14-2	104
T3B.14-3	104
T3B.17-1	104
F3B.1-1	61
F3B.1-2	61
F3B.1-3	61
F3B.1-4	61
F3B.1-5	61

(continued)

Page	Amendment No.
F3B.8-1	61
F3B.8-2	61
F3B.8-3	61
F3B.8-4	61
F3B.8-5	61
F3B.9-1	61
F3B.9-2	61
F3B.9-3	61
F3B.9-4	61
F3B.10-1	61
F3B.10-2	61
F3B.10-3	61
F3B.10-4	61
F3B.10-5	61
F3B.10-6	61
F3B.12-1	61
F3B.12-2	61
F3B.13-1	61
F3B.13-2	61
F3B.13-3	61
F3B.14-1	61
Chapter 4 TOC	104
Chapter 4 LOT	104
Chapter 4 LOF	104
Section 4.1	106
T4.1-1	105
T4.1-2	107
T4.1-3	104
Section 4.2	106
F4.2-1	92
F4.2-2	92
F4.2-3	92
F4.2-4A	Deleted
F4.2-4B	Deleted
F4.2-5	Deleted
F4.2-6	Deleted
F4.2-7A	Deleted
F4.2-7B	Deleted
F4.2-8	Original
F4.2-9	92
F4.2-10	84
F4.2-11	84

Page	Amendment No.
F4.2-12	92
F4.2-13A	Deleted
F4.2-13B	Deleted
F4.2-14	Deleted
F4.2-15	92
F4.2-16	92
Section 4.3	107
T4.3-1	Deleted
T4.3-2A	Deleted
T4.3-2B	Deleted
T4.3-3	104
T4.3-4	Deleted
T4.3-5	104
T4.3-6	104
T4.3-7	Deleted
T4.3-8	Deleted
T4.3-9	Deleted
T4.3-10	Deleted
T4.3-11	Deleted
F4.3-1	92
F4.3-2	92
F4.3-3	92
F4.3-4	92
F4.3-5	92
F4.3-6	92
F4.3-7	92
F4.3-8	92
F4.3-9	92
F4.3-10	92
F4.3-11	95
F4.3-12	92
F4.3-13	92
F4.3-14	95
F4.3-15	92
F4.3-16	92
F4.3-17	92
F4.3-18	Deleted
F4.3-19	Deleted
F4.3-20	Deleted
F4.3-21	Deleted
F4.3-22	Deleted
F4.3-23A	Deleted
F4.3-23B	Deleted
F4.3-24	92

(continued)

Page	Amendment No.
F4.3-25	Original
F4.3-26	Original
F4.3-27	92
F4.3-28	9
F4.3-29	95
F4.3-30	92
F4.3-31	92
F4.3-32	92
F4.3-33	95
F4.3-34	96
F4.3-35	92
F4.3-36	Original
F4.3-37	95
F4.3-38	92
F4.3-39	92
F4.3-40	95
F4.3-41	95
F4.3-42	Deleted
F4.3-43	Deleted
F4.3-44	Deleted
F4.3-45	Deleted
Section 4.4	105
T4.4-1	105
T4.4-1B	Deleted
T4.4-2	Deleted
T4.4-3	Deleted
T4.4-4	Deleted
F4.4-1	Deleted
F4.4-2	Deleted
F4.4-3	Deleted
F4.4-4	Deleted
F4.4-5	Deleted
F4.4-6	Deleted
F4.4-7	Deleted
F4.4-8	Deleted
F4.4-9	Deleted
F4.4-10	Deleted
F4.4-11	Deleted
F4.4-12	Deleted
F4.4-13	Deleted
F4.4-14	Deleted
F4.4-15	Deleted
F4.4-16	Deleted
F4.4-17	Deleted

Page	Amendment No.
F4.4-18	Deleted
F4.4-19	Deleted
F4.4-20	84
F4.4-21	Deleted
F4.4-21A	102
F4.4-21B	102
Section 4.5	104
Section 4.6	104
Chapter 5 TOC	104
Chapter 5 LOT	104
Chapter 5 LOF	104
Section 5.1	104
T5.1-1	Deleted
T5.1-1A	104
T5.1-1B	104
F5.1-1	(See T3.2-3)
F5.1-2	67
Notes to F5.1-2	68
F5.1-3	102
Section 5.2	104
T5.2-1	104
T5.2-2	104
T5.2-3	104
T5.2-4	104
T5.2-5	104
T5.2-6	Deleted
T5.2-6A	104
T5.2-6B	104
T5.2-7	104
F5.2-1	Deleted
F5.2-2	Deleted
F5.2-3	Deleted
Section 5.3	105
T5.3-1	104
T5.3-2A	104
T5.3-2B	104
T5.3-2C	104
T5.3-2D	104
T5.3-3A	104

(continued)

Page	Amendment No.
T5.3-3B	104
T5.3-4A	104
T5.3-4B	104
T5.3-4C	104
T5.3-5	104
T5.3-6A	104
T5.3-6B	104
T5.3-7A	104
T5.3-7B	104
T5.3-8A	104
T5.3-8B	104
T5.3-9A	104
T5.3-9B	104
T5.3-10A	104
T5.3-10B	104
T5.3-11A	104
T5.3-11B	104
T5.3-12A	104
T5.3-12B	104
T5.3-13A	104
T5.3-13B	104
T5.3-14A	104
T5.3-14B	104
T5.3-15A	104
T5.3-15B	104
T5.3-16A	104
T5.3-16B	104
T5.3-17A	104
T5.3-17B	104
F5.3-1A	8
F5.3-1B	8
F5.3-2A	91
F5.3-2B	91
F5.3-3A	91
F5.3-3B	91
F5.3-4A	91
F5.3-4B	91
Section 5.4	104
T5.4-1	104
T5.4-2	104
T5.4-3	104
T5.4-4	Deleted
T5.4-4A	104
T5.4-4B	104

Page	Amendment No.
T5.4-5	104
T5.4-6	104
T5.4-7	104
T5.4-8	104
T5.4-9	104
T5.4-10	104
T5.4-11	104
T5.4-12	104
T5.4-13	104
T5.4-14	104
T5.4-15	104
T5.4-16	104
T5.4-17	104
T5.4-18	Deleted
T5.4-19	104
T5.4-19A	104
F5.4-1	Original
F5.4-2	Original
F5.4-3	Original
F5.4-4	Deleted
F5.4-4A	102
F5.4-4B	102
F5.4-5	Original
F5.4-6 (Sh. 1)	(See Table 3.2-3)
F5.4-6 (Sh. 2)	(See Table 3.2-3)
F5.4-7 (Sh. 1)	Original
F5.4-7 (Sh. 2)	17
Notes to F5.4-7 (Sh. 1)	96
Notes to F5.4-7 (Sh. 2)	February 15, 1988
Notes to F5.4-7 (Sh. 3)	93
Notes to F5.4-7 (Sh. 4)	93
F5.4-8	93
F5.4-9	93
F5.4-10	Original
F5.4-11	Original
F5.4-12	Original
F5.4-12A	8
F5.4-12B	8
F5.4-13	102
F5.4-14	Original
F5.4-15	Original
F5.4-16	66
F5.4-17	66
F5.4-18	66
F5.4-19	61

(continued)

Page	Amendment No.
F5.4-20	91
Appendix 5A	106
Chapter 6 TOC	104
Chapter 6 LOT	104
Chapter 6 LOF	105
Section 6.1N	104
T6.1N-1	104
Section 6.1B	104
T6.1B-1	104
T6.1B-2	104
T6.1B-3	104
T6.1B-4	104
Section 6.2	107
T6.2.1-1	Deleted
T6.2.1-2	107
T6.2.1-2A	107
T6.2.1-2B	Deleted
T6.2.1-2C	Deleted
T6.2.1-3	104
T6.2.1-3A	104
T6.2.1-3B	104
T6.2.1-4	104
T6.2.1-4A	Deleted
T6.2.1-4B	Deleted
T6.2.1-5	104
T6.2.1-6	104
T6.2.1-7	Deleted
T6.2.1-8	104
T6.2.1-9	104
T6.2.1-10	104
T6.2.1-11	104
T6.2.1-12	104
T6.2.1-13	Deleted
T6.2.1-14	Deleted
T6.2.1-15	Deleted
T6.2.1-16	Deleted
T6.2.1-17	Deleted
T6.2.1-18	Deleted
T6.2.1-19	Deleted
T6.2.1-20	Deleted
T6.2.1-21	Deleted
T6.2.1-22	Deleted
T6.2.1-23	Deleted
T6.2.1-24	Deleted
T6.2.1-25	Deleted
T6.2.1-26	Deleted
T6.2.1-27	Deleted
T6.2.1-28	Deleted
T6.2.1-29	Deleted
T6.2.1-30	Deleted
T6.2.1-31	Deleted
T6.2.1-32	Deleted
T6.2.1-33	Deleted
T6.2.1-34	Deleted
T6.2.1-35	Deleted
T6.2.1-36	Deleted
T6.2.1-37	Deleted
T6.2.1-38	Deleted
T6.2.1-39	104
T6.2.1-40	Deleted
T6.2.1-41	104
T6.2.1-42	104
T6.2.1-43	104
T6.2.1-44	Deleted
T6.2.1-45	Deleted
T6.2.1-46	Deleted
T6.2.1-47	Deleted
T6.2.1-48	Deleted
T6.2.1-49	104
T6.2.1-50	104
T6.2.1-50A	Deleted
T6.2.1-51	104
T6.2.1-52	Deleted
T6.2.1-53	Deleted
T6.2.1-54	104
T6.2.1-55	Deleted
T6.2.1-56	104
T6.2.1-57	104
T6.2.1-58	Deleted
T6.2.1-59	Deleted
T6.2.1-60	Deleted
T6.2.1-61	Deleted
T6.2.1-62	Deleted
T6.2.1-63	Deleted
T6.2.1-64	Deleted

(continued)

Page	Amendment No.	Page	Amendment No.
T6.2.1-65	Deleted	T6.2.2-5	104
T6.2.1-66	Deleted	T6.2.2-6	Deleted
T6.2.1-67	Deleted	T6.2.4-1	104
T6.2.1-68	Deleted	T6.2.4-2	104
T6.2.1-69	Deleted	T6.2.4-2	104
T6.2.1-70	Deleted	T6.2.4-3	107
T6.2.1-71	Deleted	T6.2.4-4	104
T6.2.1-72	Deleted	T6.2.4-5	Deleted
T6.2.1-73	Deleted	T6.2.4-6	104
T6.2.1-74	Deleted	T6.2.5-1	Deleted
T6.2.1-75	Deleted	T6.2.5-2	Deleted
T6.2.1-76	Deleted	T6.2.5-3	Deleted
T6.2.1-77	Deleted	T6.2.5-4	Deleted
T6.2.1-78	104	T6.2.5-5	104
T6.2.1-79	Deleted	T6.2.5-6	104
T6.2.1-80	Deleted	T6.2.5A-1	Deleted
T6.2.1-81	Deleted	T6.2.5A-2	Deleted
T6.2.1-82	Deleted	T6.2.5A-3	Deleted
T6.2.1-83	Deleted	T6.2.5A-4	Deleted
T6.2.1-84	Deleted	T6.2.5A-5	Deleted
T6.2.1-85	Deleted	T6.2.5A-6	Deleted
T6.2.1-86	Deleted	F6.2.1-1	Deleted
T6.2.1-87	Deleted	F6.2.1-2	Deleted
T6.2.1-88	Deleted	F6.2.1-3	103
T6.2.1-89	Deleted	F6.2.1-4	103
T6.2.1-90	Deleted	F6.2.1-5	69
T6.2.1-91	Deleted	F6.2.1-6	69
T6.2.1-92	104	F6.2.1-7	69
T6.2.1-93	Deleted	F6.2.1-8	69
T6.2.1-94	104	F6.2.1-9	102
T6.2.1-95	Deleted	F6.2.1-10	102
T6.2.1-96	Deleted	F6.2.1-11	102
T6.2.1-97	Deleted	F6.2.1-12	102
T6.2.1-98	Deleted	F6.2.1-13	102
T6.2.1-99	Deleted	F6.2.1-14	102
T6.2.1-100	Deleted	F6.2.1-15	102
T6.2.1-101	Deleted	F6.2.1-16	102
T6.2.1-102	Deleted	F6.2.1-16A	69
T6.2.1-103	Deleted	F6.2.1-17	93
T6.2.1-104	Deleted	F6.2.1-18	69
T6.2.1-105	Deleted	F6.2.1-19	69
T6.2.2-1	104	F6.2.1-20	Deleted
T6.2.2-2	104	F6.2.1-21	78
T6.2.2-3	104	F6.2.1-22	78
T6.2.2-4	104	F6.2.1-23	78

(continued)

Page	Amendment No.
F6.2.1-24	78
F6.2.1-25	78
F6.2.1-26	78
F6.2.1-27	78
F6.2.1-28	78
F6.2.1-29	78
F6.2.1-30	Deleted
F6.2.1-31	Deleted
F6.2.1-32	Deleted
F6.2.1-33	Deleted
F6.2.1-34	Deleted
F6.2.1-35	Deleted
F6.2.1-36	Deleted
F6.2.1-37	Deleted
F6.2.1-38	Deleted
F6.2.1-39	Deleted
F6.2.1-40	Deleted
F6.2.1-41	Deleted
F6.2.1-42	Deleted
F6.2.1-43	Deleted
F6.2.1-44	Deleted
F6.2.1-45	Deleted
F6.2.1-46	10
F6.2.1-47	10
F6.2.1-48	10
F6.2.1-49	Deleted
F6.2.1-50	Deleted
F6.2.1-51	Deleted
F6.2.1-52	Deleted
F6.2.1-53	78
F6.2.1-54	Deleted
F6.2.1-55	Deleted
F6.2.1-56	Deleted
F6.2.1-57	78
F6.2.1-58	Deleted
F6.2.1-59	Deleted
F6.2.1-60	Deleted
F6.2.1-61	Deleted
F6.2.1-62	Deleted
F6.2.1-63	Deleted
F6.2.1-64	Deleted
F6.2.1-65	Deleted
F6.2.1-66	Deleted
F6.2.1-67	Deleted
F6.2.1-68	Deleted

Page	Amendment No.
F6.2.1-69	Deleted
F6.2.1-70	Deleted
F6.2.2-1	94
F6.2.2-2	96
F6.2.2-3	102
F6.2.2-3A	102
F6.2.2-4	102
F6.2.2-5	102
F6.2.4-1 (Sh. 1)	87
F6.2.4-1 (Sh. 2)	83
F6.2.4-1 (Sh. 3)	104
F6.2.4-1 (Sh. 4)	99
F6.2.4-1 (Sh. 5)	96
F6.2.4-1 (Sh. 6)	66
F6.2.4-1 (Sh. 7)	66
F6.2.4-1 (Sh. 8)	83
F6.2.4-1 (Sh. 9)	102
F6.2.4-1 (Sh. 10)	96
F6.2.4-1 (Sh. 11)	96
F6.2.4-1 (Sh. 12)	104
F6.2.5-1	102
F6.2.5-2	Deleted
F6.2.5-3A	36
F6.2.5A-1	Deleted
F6.2.5A-2	Deleted
F6.2.5A-3	Deleted
F6.2.5A-4	Deleted
F6.2.5A-5	Deleted
F6.2.5A-6	Deleted
F6.2.5A-7	Deleted
F6.2.5A-8	Deleted
F6.2.5A-9	Deleted
F6.2.6-1	13
Section 6.3	107
T6.3-1	106
T6.3-2	104
T6.3-3	107
T6.3-4	104
T6.3-5	104
T6.3-6	104
T6.3-7	107
T6.3-8	104
T6.3-9	104
T6.3-10	104

(continued)

Page	Amendment No.
T6.3-11	106
F6.3-2 (Sh. 1)	78
F6.3-2 (Sh. 2)	16
Note F6.3-2 (Sh. 1)	Original
Note F6.3-2 (Sh. 2)	Original
Note F6.3-2 (Sh. 3)	100
Note F6.3-2 (Sh. 4)	78
Note F6.3-2 (Sh. 5)	96
Note F6.3-2 (Sh. 6)	96
Note F6.3-2 (Sh. 7)	96
Note F6.3-2 (Sh. 8)	96
Note F6.3-2 (Sh. 9)	96
Note F6.3-2 (Sh. 10)	96
Note F6.3-2 (Sh. 11)	96
Note F6.3-2 (Sh. 12)	96
Note F6.3-2 (Sh. 13)	96
Note F6.3-2 (Sh. 14)	96
Note F6.3-2 (Sh. 15)	96
Note F6.3-2 (Sh. 16)	96
F6.3-3	53
F6.3-4	53
F6.3-5	53
F6.3-6	52
F6.3-7	6
F6.3-8	96
F6.3-9	96
Section 6.4	104
T6.4-1	104
T6.4-2	Deleted
T6.4-3	104
T6.4-4	104
F6.4-1	Original
F6.4-2	Original
F6.4-3	13
Section 6.5	104
T6.5-1	107
T6.5-2	104
T6.5-3	104
T6.5-4	104
T6.5-5	104
T6.5-6	104
T6.5-7	104
F6.5-1	Original
F6.5-2	96
F6.5-3	Original
F6.5-4 (Sh. 1)	9
F6.5-4 (Sh. 2)	9
F6.5-4 (Sh. 3)	96
F6.5-4 (Sh. 4)	9
F6.5-4 (Sh. 5)	9
Section 6.6	104
Section 6.7	104
Chapter 7 TOC	104
Chapter 7 LOT	104
Chapter 7 LOF	104
Section 7.1	104
T7.1-1	104
T7.1-2.2	104
T7.1-2.3	104
T7.1-2.4	104
T7.1-2.5	104
T7.1-2.6	104
T7.1-2.7	104
T7.1-3	104
F7.1-1	Original
F7.1-2	5
F7.1-3 (Sh. 1)	66
F7.1-3 (Sh. 2)	78
F7.1-3 (Sh. 3)	66
F7.1-3 (Sh. 4)	66
F7.1-3 (Sh. 5)	97
F7.1-3 (Sh. 5A)	92
F7.1-3 (Sh. 6)	66
F7.1-3 (Sh. 7)	78
F7.1-3 (Sh. 8)	66
F7.1-3 (Sh. 9)	94
F7.1-3 (Sh. 10)	87
F7.1-3 (Sh. 11)	87
F7.1-3 (Sh. 12)	87
F7.1-3 (Sh. 13)	99
F7.1-3 (Sh. 14)	66
F7.1-3 (Sh. 15)	96
F7.1-3 (Sh. 16)	96
F7.1-3 (Sh. 17)	66
F7.1-3 (Sh. 18)	78

(continued)

Page	Amendment No.
F7.1-3 (Sh. 19)	87
F7.1-3 (Sh. 20)	66
F7.1-3 (Sh. 21)	66
F7.1-3 (Sh. 22)	66
F7.1-3 (SH. 23)	87
F7.1-3 (SH. 23A)	100
F7-1.3 (Sh. 24)	87
F7.1-3 (Sh. 25)	87
F7.1-3 (SH. 26)	94
F7.1-3 (SH. 26A)	87
F7.1-3 (SH. 27)	87
F7.1-3 (SH. 28)	87
F7.1-3 (SH. 29)	87
F7.1-3 (SH. 30)	87
F7.1-3 (SH. 31)	96
F7.1-3 (SH. 32)	96
F7.1-3 (SH. 33)	87
F7.1-3 (SH. 34)	87
F7.1-4	86
Section 7.2	105
T7.2-1	104
T7.2-2	104
T7.2-3	104
T7.2-4	104
F7.2-1 (Sh. 1)	76
F7.2-1 (Sh. 2)	77
F7.2-1 (Sh. 3)	94
F7.2-1 (Sh. 4)	100
F7.2-1 (Sh. 5)	100
F7.2-1 (Sh. 6)	76
F7.2-1 (Sh. 7)	87
F7.2-1 (Sh. 8)	76
F7.2-1 (Sh. 9)	96
F7.2-1 (Sh. 10)	96
F7.2-1 (Sh. 11)	96
F7.2-1 (Sh. 12)	76
F7.2-1 (Sh. 13)	96
F7.2-1 (Sh. 14)	76
F7.2-1 (Sh. 15)	79
F7.2-1 (Sh. 16)	96
F7.2-1 (Sh. 17)	76
F7.2-1 (Sh. 18)	76
F7.2-2	76
F7.2-3	Original

Page	Amendment No.
F7.2-4	91
F7.2-5	91
F7.2-6	91
Section 7.3	104
T7.3-1	104
T7.3-2	104
T7.3-3	104
T7.3-4	104
T7.3-5	104
T7.3-6	104
F7.3-1	October 8, 1980
F7.3-2	96
F7.3-3	11
F7.3-4	106
Section 7.4	104
T7.4-1	104
T7.4-2	104
T7 4-3	104
Section 7.5	105
T7.5-1	104
T7.5-2	105
T7.5-3	104
T7.5-4	104
T7.5-5	104
7.5-6	104
T7.5-7	Deleted
T7.5-7A	105
T7.5-7B	105
T7.5-7C	105
T7.5-7D	105
T7.5-7E	105
T7.5-7F	104
Section 7.6	104
T7.6-1	104
T7.6-2	104
T7.6-3	104
T7.6-4	104
T7.6-5	104
T7.6-6	104
F7.6-1	12
F7.6-2 (Sh. 1)	83

(continued)

Page	Amendment No.
F7.6-2 (Sh. 2)	83
F7.6-3	Original
F7.6-4 (Sh. 1)	Original
F7.6-4 (Sh. 2)	Original
F7.6-5	67
F7.6-6	15
Section 7.7	106
T7.7-1	104
T7.7-2	Deleted
F7.7-1	96
F7.7-2	November 30, 1979
F7.7-3	Original
F7.7-4	Original
F7.7-5	96
F7.7-6	96
F7.7-7	Original
F7.7-8	96
F7.7-9	Original
F7.7-10	95
F7.7-11	95
F7.7-12	95
F7.7-13	95
F7.7-14	102
F7.7-14A	100
F7.7-15	Original
F7.7-16	2
F7.7-17	96
F7.7-18	78
Section 7.8	104
F7.8-1	96
Chapter 8 TOC	104
Chapter 8 LOT	104
Chapter 8 LOF	106
Section 8.1	104
T8.1-1	104
Section 8.2	106
T8.2-1	Deleted
F8.2-1	106
F8.2-2	83
F8.2-4	106
F8.2-5 (Sheet 1)	106
F8.2-5 (Sheet 2)	104
F8.2-6	83
F8.2-7 (Sheet 1)	106
F8.2-7 (Sheet 2)	106
F8.2-8	95
F8.2-8a	95
F8.2-9	106
F8.2-10 (Sheet 1)	97
F8.2-10 (Sheet 2)	97
F8.2-11	106
F8.2-11A	106
F8.2-12	104
Section 8.3	106
T8.3-1	104
T8.3-1A	Deleted
T8.3-1B	Deleted
T8.3-1C	104
T8.3-1D	Deleted
T8.3-2	104
T8.3-3	104
T8.3-4	104
T8.3-4A	Deleted
T8.3-4B	Deleted
T8.3-4C	Deleted
T8.3-5	Deleted
T8.3-6	Deleted
T8.3-7	104
T8.3-8	Deleted
T8.3-9	Deleted
T8.3-10	104
T8.3-11	104
F8.3-1	96
F8.3-2	96
F8.3-3	(see T3.2-3)
F8.3-4	(see T3.2-3)
F8.3-5	(see T3.2-3)
F8.3-6	(see T3.2-3)
F8.3-7	(see T3.2-3)
F8.3-8	(see T3.2-3)
F8.3-9	(see T3.2-3)
F8.3-10	(see T3.2-3)
F8.3-11	(see T3.2-3)
F8.3-12	(see T3.2-3)

(continued)

Page	Amendment No.	Page	Amendment No.
F8.3-13	(see T3.2-3)		
F8.3-14	(see T3.2-3)	Section 9.1	107
F8.3-14A	(see T3.2-3)	T9.1-1	104
F8.3-15	(see T3.2-3)	T9.1-2	104
F8.3-15A	(see T3.2-3)	T9.1-3	Deleted
F8.3-15B	(see T3.2-3)	T9.1-4	107
F8.3-15C	(see T3.2-3)	F9.1-1	Original
F8.3-16	101	F9.1-2	98
F8.3-17	9	F9.1-2A	98
F8.3-18	(see T3.2-3)	F9.1-2B	98
F8.3-19	91	F9.1-3	Original
F8.3-20	Deleted	F9.1-4	104
F8.3-21	Deleted	F9.1-5	Original
F8.3-22	Deleted	F9.1-6	66
F8.3-23	Deleted	F9.1-7	January 30, 1981
F8.3-24	Deleted	F9.1-8	Original
F8.3-25	Deleted	F9.1-9	Original
F8.3-26	Deleted	F9.1-10	13
F8.3-27	Deleted	F9.1-11	Original
F8.3-28	Deleted	F9.1-12	87
F8.3-29	Deleted	F9.1-13	(See T3.2-3)
F8.3-30	Deleted	F9.1-14 (Sh. 1)	October 8, 1980
F8.3-31	Deleted	F9.1-14 (Sh. 2)	October 8, 1980
F8.3-32	Deleted	F9.1-14 (Sh. 3)	October 8, 1980
F8.3-33	Deleted	F9.1-15 (Sh. 1)	October 8, 1980
F8.3-34	Deleted	F9.1-15 (Sh. 2)	October 8, 1980
F8.3-35	Deleted		
F8.3-36	Deleted	Section 9.2	107
F8.3-37	Deleted	T9.2-1	104
F8.3-38	Deleted	T9.2-2	104
F8.3-39	Deleted	T9.2-3	104
F8.3-40	Deleted	T9.2-4	104
F8.3-41	Deleted	T9.2-5	104
F8.3-42	Deleted	T9.2-6	Deleted
F8.3-43	Deleted	T9.2-7	Deleted
F8.3-44	Deleted	T9.2-8	104
F8.3-45	Deleted	T9.2-9	104
		T9.2-10	104
Appendix 8A	104	T9.2-11	Deleted
		T9.2-12	Deleted
Appendix 8B	106	T9.2-13	104
		T9.2-14	Deleted
Chapter 9 TOC	104	F9.2-1	94
Chapter 9 LOT	104	F9.2-2	68
Chapter 9 (LOF)	107	F9.2-3	(See T3.2-3)

(continued)

Page	Amendment No.
F9.2-4	(See T3.2-3)
F9.2-4A	(See T3.2-3)
F9.2-5	(See T3.2-3)
F9.2-6	(See T3.2-3)
F9.2-7	(See T3.2-3)
F9.2-8	68
F9.2-9	10
F9.2-10	68
F9.2-11	63
F9.2-12	July 31, 1980
F9.2-13	July 31, 1980
F9.2-14	July 31, 1980
F9.2-15	(See T3.2-3)
F9.2-16	(See T3.2-3)
Section 9.3	107
T9.3-1	106
T9.3-2	104
T9.3-3	104
T9.3-4	104
T9.3-5	104
T9.3-6	104
T9.3-7	107
T9.3-8	104
T9.3-9	106
F9.3-1	(See T3.2-3)
F9.3-2	(See T3.2-3)
F9.3-3	Deleted
F9.3-4	(See T3.2-3)
F9.3-5	(See T3.2-3)
F9.3-6	(See T3.2-3)
F9.3-7	(See T3.2-3)
F9.3-8	(See T3.2-3)
F9.3-9	(See T3.2-3)
F9.3-10	(See T3.2-3)
F9.3-11	(See T3.2-3)
Section 9.4	104
T9.4-1	104
T9.4-2	105
T9.4-3	104
T9.4-4	104
T9.4-5	104
T9.4-6	104
T9.4-7	104
T9.4-8	104
T9.4-9	104
T9.4-10	104
F9.4-1	(See T3.2-3)
F9.4-2	(See T3.2-3)
F9.4-3	(See T3.2-3)
F9.4-4	(See T3.2-3)
F9.4-5	(See T3.2-3)
F9.4-6	(See T3.2-3)
F9.4-7	(See T3.2-3)
F9.4-8	(See T3.2-3)
F9.4-9	(See T3.2-3)
F9.4-10	Original
F9.4-11	(See T3.2-3)
F9.4-12	(See T3.2-3)
F9.4-13	96
F9.4-14	(See T3.2-3)
F9.4-15	(See T3.2-3)
Section 9.4A	104
Section 9.4B	104
Section 9.4C	105
Section 9.4D	104
Section 9.4E	105
Section 9.4F	104
Section 9.5	105
T9.5-1	Deleted
T9.5-2	Deleted
T9.5-3	Deleted
T9.5-4	Deleted
T9.5-5	Deleted
T9.5-6	Deleted
T9.5-7	Deleted
T9.5-8	Deleted
T9.5-9	Deleted
T9.5-10	Deleted
T9.5-11	104
T9.5-12	104
T9.5-13	104

(continued)

Page	Amendment No.
T9.5-14	104
T9.5-15	Deleted
T9.5-16	104
T9.5-17	104
T9.5-18	104
F9.5-1	Deleted
F9.5-2	Deleted
F9.5-3	Deleted
F9.5-4	Deleted
F9.5-5	Deleted
F9.5-6	Deleted
F9.5-7	Deleted
F9.5-8	Deleted
F9.5-9	Deleted
F9.5-10	Deleted
F9.5-11	Deleted
F9.5-12	Deleted
F9.5-13	Deleted
F9.5-14	Deleted
F9.5-15	Deleted
F9.5-16	Deleted
F9.5-17	Deleted
F9.5-18	Deleted
F9.5-19	Deleted
F9.5-20	Deleted
F9.5-21	Deleted
F9.5-22	Deleted
F9.5-23	Deleted
F9.5-24	Deleted
F9.5-25	Deleted
F9.5-26	Deleted
F9.5-27	Deleted
F9.5-28	Deleted
F9.5-29	Deleted
F9.5-30	Deleted
F9.5-31	Deleted
F9.5-32	Deleted
F9.5-33	Deleted
F9.5-34	Deleted
F9.5-35	Deleted
F9.5-36	Deleted
F9.5-37	Deleted
F9.5-38	Deleted
F9.5-39	Deleted
F9.5-40	Deleted

Page	Amendment No.
F9.5-41	Deleted
F9.5-42	Deleted
F9.5-43	94 (See T3.2-3)
F9.5-44	94 (See T3.2-3)
F9.5-44A	94 (See T3.2-3)
F9.5-45	94 (see T3.2-3)
F9.5-46	94 (See T3.2-3)
F9.5-47	94 (see T3.2-3)
F9.5-48	94 (See T3.2-3)
F9.5-48A	94 (See T3.2-3)
F9.5-49	Deleted
F9.5-50	Deleted
F9.5-51	104
F9.5-52	94 (See T3.2-3)
F9.5-53	June 15, 1978
F9.5-54	94 (See T3.2-3)
F9.5-55	94 (See T3.2-3)
F9.5-56	94 (See T3.2-3)
F9.5-57	94 (See T3.2-3)
F9.5-58	11
F9.5-59	11
F9.5-60	96
F9.5-61	94 (See T3.2-3)
F9.5-62	94 (See T3.2-3)
Chapter 10 TOC	104
Chapter 10 LOT	104
Chapter 10 LOF	104
Section 10.1	104
T10.1-1	104
F10.1-1	103
F10.1-2	103
Section 10.2	106
F10.2-1	98
Section 10.3	104
T10.3-1	104
T10.3-2	104
T10.3-3	104
T10.3-4	104
T10.3-5	104
T10.3-6	104
T10.3-7	104

(continued)

Page	Amendment No.
T10.3-8	104
T10.3-9	104
T10.3-10	Deleted
T10.3-11	104
F10.3-1	94
Section 10.4	107
T10.4-1	104
T10.4-2	104
T10.4-3A	104
T10.4-4	Deleted
T10.4-5	104
T10.4-6	104
T10.4-7	104
T10.4-8	104
T10.4-9	104
T10.4-10	104
T10.4-11	104
T10.4-12	104
T10.4-13	104
T10.4-14	104
T10.4-15	Deleted
T10.4-16	104
T10.4-17	104
T10.4-18	104
T10.4-19	104
T10.4-20	104
F10.4-1	Original
F10.4-2	Original
F10.4-3	(See T3.2-3)
F10.4-4	(See T3.2-3)
F10.4-5	(See T3.2-3)
F10.4-6	(See T3.2-3)
F10.4-7	(See T3.2-3)
F10.4-8	(See T3.2-3)
F10.4-9	(See T3.2-3)
F10.4-10	(See T3.2-3)
F10.4-11	(See T3.2-3)
F10.4-12	102
F10.4-13	94 (See T3.2-3)
F10.4-14	94 (See T3.2-3)
F10.4-15	94 (See T3.2-3)
F10.4-16	94 (See T3.2-3)
F10.4-17	94 (See T3.2-3)
F10.4-18	94 (See T3.2-3)
F10.4-19	94 (See T3.2-3)
F10.4-20	94 (See T3.2-3)
F10.4-21	6
F10.4-22	Deleted
F10.4-22A	102
F10.4-22B	102
F10.4-23	December 15, 1980
Chapter 11 TOC	105
Chapter 11 LOT	104
Chapter 11 LOF	107
Section 11.1	104
T11.1-1	104
T11.1-2	Deleted
T11.1-3	104
T11.1-4	104
T11.1-5	104
T11.1-6	Deleted
T11.1-7	104
Section 11.2	104
T11.2-1	104
T11.2-2	104
T11.2-3	104
T11.2-4	104
T11.2-5	104
T11.2-6	104
T11.2-7	Deleted
T11.2-8	104
T11.2-9	104
T11.2-10	104
F11.2-1	77
F11.2-2	96
F11.2-3	96
F11.2-4	96
F11.2-5	96
F11.2-6	96
F11.2-7	96
F11.2-8	96
F11.2-9	96
Section 11.3	104
T11.3-1	104
T11.3-2	104

(continued)

Page	Amendment No.
T11.3-3	104
T11.3-4	104
T11.3-5	104
T11.3-6	104
T11.3-7	104
F11.3-1	(See T3.2-3)
F11.3-2	96
F11.3-3	96
F11.3-4	Original
F11.3-5	76
Section 11.4	105
T11.4-1	104
T11.4-2	Deleted
F11.4-1	January 21, 1985
F11.4-2	86
Section 11.5	105
T11.5-1	104
T11.5-2	104
T11.5-3	104
T11.5-4	104
F11.5-1	105
Appendix 11A	104
T11A-1	104
T11A-2	104
T11A-3	104
T11A-4	104
T11A-5	104
T11A-6	104
F11A-1	Original
F11A-2	85
F11A-3	85
Chapter 12 TOC	105
Section 12.1	105
F12.1-1	76
Section 12.2	105
T12.2-1	104
T12.2-2	104
T12.2-3	104
T12.2-4	104

Page	Amendment No.
T12.2-5	104
T12.2-6	104
T12.2-7	104
T12.2-8	104
T12.2-9	104
T12.2-10	104
T12.2-11	104
T12.2-12	104
T12.2-13	104
T12.2-14	104
T12.2-15	104
T12.2-16	104
T12.2-17	104
T12.2-17A	104
T12.2-18	Deleted
T12.2-18A	Deleted
T12.2-19	104
T12.2-19A	104
T12.2-19BA	104
T12.2-19C	104
T12.2-19D	104
T12.2-19E	104
T12.2-20	104
T12.2-21	104
T12.2-22	104
T12.2-23	104
T12.2-24	104
T12.2-25	104
T12.2-26	104
Section 12.3	104
T12.3-1	104
T12.3-2	104
T12.3-3	104
T12.3-4	104
T12.3-5	104
T12.3-6	104
T12.3-7	104
T12.3-8	104
T12.3-9	104
F12.3-1	87
F12.3-2	87
F12.3-3	87
F12.3-4	76
F12.3-5	96

(continued)

Page	Amendment No.
F12.3-5.1	76
F12.3-6	99
F12.3-7	99
F12.3-8	100
F12.3-9	98
F12.3-10	99
F12.3-11	76
F12.3-12	99
F12.3-13	99
F12.3-14	99
F12.3-14.1	76
F12.3-15	99
F12.3-16	99
F12.3-17	99
F12.3-18	99
F12.3-19	99
F12.3-20	96
F12.3-21	98
F12.3-21.1	96
F12.3-21.2	96
F12.3-22	83
F12.3-22.1	96
F12.3-22.2	96
F12.3-23	99
F12.3-23.1	99
F12.3-23.2	96
F12.3-23.3	86
F12.3-23.4	99
F12.3-23.5	99
F12.3-23.6	100
F12.3-23.7	98
F12.3-23.8	99
F12.3-24	Original
F12.3-25	Original
F12.3-26	Original
F12.3-27	Original
F12.3-28	Original
F12.3-29	Original
F12.3-30	Original
F12.3-31	Original
F12.3-32	Original
F12.3-33 (Sh.1)	83
F12.3-33 (Sh. 2)	83
F12.3-33 (Sh. 3)	102

Page	Amendment No.
Section 12.4	104
T12.4-1	104
T12.4-2	104
T12.4-3	104
T12.4-4	104
T12.4-5	104
T12.4-6	104
T12.4-7	104
T12.4-8	104
T12.4-9	104
T12.4-10	104
T12.4-11	104
F12.4-1	Original
Section 12.5	105
T12.5-1	106
T12.5-2	Deleted
F12.5-1	87
F12.5-2	83
F12.5-3	87
Chapter 13 TOC	107
Chapter 13 LOT	104
Chapter 13 LOF	104
Section 13.1	107
T13.1-1	107
T13.1-2	104
F13.1-1	102
F13.1-2	107
F13.1-3	106
F13.1-4	November 3, 1986
Appendix 13.1A	107
Section 13.2	104
F13.2-1	41
Section 13.3	104
Section 13.4	105
Section 13.5	104
T13.5-1	104
T13.5-2	104

(continued)

Page	Amendment No.
T13.5-3	104
T13.5-4	104
T13.5-5	104
T13.5-6	104
T13.5-7	Deleted
T13.5-8	104
T13.5-9	104
F13.5-1	100
Section 13.6	107
Chapter 14 TOC	104
Section 14.1	104
Section 14.2	105
T14.2-1	Deleted
T14.2-2	104
T14.2-3	104
F14.2-1	83
F14.2-2	87
F14.2-3 (Sh. 1)	88
F14.2-3 (Sh. 2)	87
F14.2-4A	88
F14.2-4B	88
Chapter 15 TOC	104
Chapter 15 LOT	104
Chapter 15 LOF	104
Section 15.0	104
T15.0-1	104
T15.0-2	Deleted
T15.0-3	Deleted
T15.0-4	107
T15.0-5	Deleted
T15.0-6	104
T15.0-7	104
T15.0-8	104
F15.0-1	73
F15.0-2A	August 1, 1996
F15.0-2B	August 1, 1996
F15.0-3	Original
F15.0-4	5
F15.0-5	5

Page	Amendment No.
F15.0-6	84
F15.0-7	February 20, 1981
F15.0-8	15
F15.0-9	15
F15.0-10	78
F15.0-11	78
F15.0-12	15
F15.0-13	15
F15.0-14	15
F15.0-15	March 31, 1981
F15.0-16	15
F15.0-17	94
F15.0-18	15
F15.0-19	78
F15.0-20	92
F15.0-21	54
F15.0-22	15
F15.0-23	15
F15.0-24	70
F15.0-25	February 20, 1981
F15.0-26	76
F15.0-27	76
F15.0-28	February 20, 1981
F15.0-29	76
F15.0-30	76
F15.0-31	87
Section 15.1	105
T15.1-1	Deleted
T15.1-2	104
T15.1-3	104
T15.1-4	104
T15.1-4A	Deleted
F15.1-1A	Deleted
F15.1-1B	Deleted
F15.1-2	Deleted
F15.1-3	Deleted
F15.1-4	Deleted
F15.1-5	Deleted
F15.1-6	Deleted
F15.1-7	Deleted
F15.1-8	Deleted
F15.1-9	Deleted
F15.1-10	Deleted
F15.1-11	Deleted

(continued)

Page	Amendment No.
F15.1-12	Deleted
F15.1-13	Deleted
F15.1-14	Deleted
F15.1-15	Deleted
F15.1-16	Deleted
F15.1-17	Deleted
F15.1-18	Deleted
F15.1-19	Deleted
F15.1-20	Deleted
Section 15.2	104
T15.2-1	Deleted
T15.2-2	Deleted
F15.2-1A	Deleted
F15.2-1B	Deleted
F15.2-2	Deleted
F15.2-3	Deleted
F15.2-4	Deleted
F15.2-5	Deleted
F15.2-6	Deleted
F15.2-7	Deleted
F15.2-8	Deleted
F15.2-9	Deleted
F15.2-10	Deleted
F15.2-11	Deleted
F15.2-12	Deleted
F15.2-13	Deleted
F15.2-14	Deleted
F15.2-15	Deleted
F15.2-16	Deleted
F15.2-17	Deleted
F15.2-18	Deleted
F15.2-19	Deleted
F15.2-20	Deleted
F15.2-21	Deleted
F15.2-22	Deleted
F15.2-23	Deleted
F15.2-24	Deleted
F15.2-25	Deleted
F15.2-26A	Deleted
Section 15.3	104
T15.3-1	104
T15.3-2	Deleted
F15.3-1A	Deleted

Page	Amendment No.
F15.3-B	Deleted
F15.3-2	Deleted
F15.3-3	Deleted
F15.3-4	Deleted
F15.3-5	Deleted
F15.3-6	Deleted
F15.3-7	Deleted
F15.3-8	Deleted
F15.3-9	Deleted
F15.3-10	Deleted
F15.3-11	Deleted
F15.3-12	Deleted
F15.3-13	Deleted
F15.3-14	Deleted
F15.3-15	Deleted
F15.3-16	Deleted
F15.3-17	Deleted
F15.3-18	Deleted
F15.3-19	Deleted
F15.3-20	Deleted
F15.3-21	Deleted
F15.3-22	Deleted
F15.3-23	Deleted
F15.3-24	Deleted
Section 15.4	106
T15.4-1	Deleted
T15.4-2	Deleted
T15.4-3	Deleted
T15.4-4	104
F15.4-1A	Deleted
F15.4-1B	Deleted
F15.4-2	Deleted
F15.4-3	Deleted
F15.4-4	Deleted
F15.4-5	Deleted
F15.4-6	Deleted
F15.4-7	Deleted
F15.4-8	Deleted
F15.4-9	Deleted
F15.4-10	Deleted
F15.4-11	Deleted
F15.4-12	Deleted
F15.4-13	Deleted
F15.4-14	Deleted

(continued)

Page	Amendment No.
F15.4-15	Deleted
F15.4-16	Deleted
F15.4-17	Deleted
F15.4-18	Deleted
F15.4-19	Deleted
F15.4-20	Deleted
F15.4-21	Deleted
F15.4-22	Deleted
F15.4-23	Deleted
F15.4-24	Deleted
F15.4-25	Deleted
F15.4-26	Deleted
F15.4-27	Deleted
F15.4-28	Deleted
F15.4-29	Deleted
F15.4-30	Deleted
Section 15.5	105
T15.5-1	104
F15.5-1	Deleted
F15.5-2	Deleted
F15.5-3	Deleted
Section 15.6	105
T15.6-1	104
T15.6-2	104
T15.6-3	104
T15.6-4	105
T15.6-5	105
T15.6-6	104
T15.6-7	104
T15.6-8	106
T15.6-9	105
T15.6-10	106
T15.6-11	105
T15.6-12	104
T15.6-20	104
T15.6-21	104
F15.6-1A	Deleted
F15.6-1B	Deleted
F15.6-2	Deleted
F15.6-3	Deleted
F15.6-4	Deleted
F15.6-5	95
F15.6-6	95

Page	Amendment No.
F15.6-7A	Deleted
F15.6-7B	Deleted
F15.6-7C	Deleted
F15.6-7D	Deleted
F15.6-8	Deleted
F15.6-9	Deleted
F15.6-10	Deleted
F15.6-11	Deleted
F15.6-12	Deleted
F15.6-13	Deleted
F15.6-14	Deleted
F15.6-15	Deleted
F15.6-16	Deleted
F15.6-17	Deleted
F15.6-18	Deleted
F15.6-19	Deleted
F15.6-20	Deleted
F15.6-21	Deleted
F15.6-22	Deleted
F15.6-23	Deleted
F15.6-24	Deleted
F15.6-25	Deleted
F15.6-26	Deleted
F15.6-27	Deleted
F15.6-28	Deleted
F15.6-29	Deleted
F15.6-30	Deleted
F15.6-31	Deleted
F15.6-32	Deleted
F15.6-33	Deleted
F15.6-34	Deleted
F15.6-35	Deleted
F15.6-36	Deleted
F15.6-37	Deleted
F15.6-38	Deleted
F15.6-39	Deleted
F15.6-40	Deleted
F15.6-41	Deleted
F15.6-42	Deleted
F15.6-43	Deleted
F15.6-44	Deleted
F15.6-45	Deleted
F15.6-46	Deleted
F15.6-47	Deleted
F15.6-48	Deleted

(continued)

Page	Amendment No.
F15.6-49	Deleted
F15.6-50	Deleted
F15.6-51	Deleted
F15.6-52	Deleted
F15.6-53	Deleted
F15.6-54	Deleted
F15.6-55	Deleted
F15.6-56	Deleted
F15.6-57	Deleted
F15.6-58	Deleted
F15.6-59	Deleted
F15.6-60	Deleted
F15.6-61	Deleted
F15.6-62	Deleted
F15.6-63	Deleted
F15.6-64	Deleted
F15.6-65	Deleted
F15.6-66	Deleted
F15.6-67	Deleted
F15.6-68	Deleted
F15.6-69	Deleted
F15.6-70	Deleted
F15.6-71	Deleted
F15.6-72	Deleted
F15.6-73	Deleted
F15.6-74	Deleted
F15.6-75	Deleted
F15.6-76	Deleted
F15.6-77	Deleted
F15.6-78	Deleted
F15.6-79	Deleted
F15.6-80	Deleted
F15.6-81	Deleted
F15.6-82	Deleted
F15.6-83	Deleted
F15.6-84	Deleted
F15.6-85	Deleted
F15.6-86	Deleted
F15.6-87	Deleted
F15.6-88	Deleted
F15.6-89	Deleted
Section 15.7	104
T15.7-1	104
T15.7-2	104

Page	Amendment No.
T15.7-3	104
T15.7-4	104
T15.7-5	Deleted
T15.7-6	104
T15.7-7	104
Section 15.8	104
Appendix 15A	Deleted
Appendix 15B	104
T15B-1	104
Chapter 16 TOC	104
Section 16.1	104
Section 16.2	104
Chapter 17 TOC	106
Chapter 17 LOT	104
Chapter 17 LOF	104
Section 17.1	104
Section 17.2	107
T17.2-1	104
T17.2-2	104
Appendix 17A	106
T17A-1	107
T17A-2	107
RESPONSE TO THE NRC ACTION PLAN	
Chapter TMI TOC	104
Section I.A	104
Section I.B	106
Section I.C	104
Section I.D	104

(continued)

Page	Amendment No.
Section I.G	104
Section II.B	104
TII.B.2-1	Deleted
TII.B.2-2	Deleted
TII.B.2-3	Deleted
TII.B.2-4	104
TII.B.2-5,	104
TII.B.2-6	104
TII.B.3-1	Deleted
FII.B.2-1 - FII.B.2-24	77
FII.B.2-25	77
FII.B.2-26	77
FII.B.2-27	77
FII.B.2-28	Deleted
FII.B.2-29	Deleted
FII.B.2-30	77
FII.B.2-31	77
FII.B.2-32	77
FII.B.2-33	77
FII.B.2-34	77
FII.B.2-35	77
FII.B.2-36 (Sh. 1)	77
FII.B.2-36 (Sh. 2)	77
FII.B.2-37	77
FII.B.2-38	93
FII.B.2-39 (Sh. 1)	77
FII.B.2-39 (Sh. 2)	77
FII.B.2-39 (Sh. 3)	96
FII.B.2-40 (Sh. 1)	77
FII.B.2-40 (Sh. 2)	77
FII.B.2-41	77
FII.B.2-42	77
FII.B.2-42.1	93
FII.B.2-42.2	93
FII.B.2.42.3	93
FII.B.2.42.4	94
FII.B.2-43	87
FII.B.2-44	87
FII.B.2-45	87
FII.B.2-46	87
FII.B.2-47	87
FII.B.2-48	Deleted
FII.B.2-49	Deleted
FII.B.2-50	87

Page	Amendment No.
FII.B.2-51	87
FII.B.2-52	87
FII.B.2-53	87
FII.B.2-54	87
FII.B.2-55	87
FII.B.2-56	87
FII.B.2-57	87
FII.B.2-58	87
FII.B.2-59	87
FII.B.2-60	87
FII.B.2-61	87
FII.B.2-62	87
FII.B.2-63	87
FII.B.2-64	87
FII.B.2-65	87
FII.B.2-66	93
FII.B.2-67	93
FII.B.2-68	87
FII.B.2-69	87
FII.B.2-70	96
FII.B.2-71	87
FII.B.2-72	87
FII.B.2-73	87
FII.B.2-74	87
FII.B.2-75	87
FII.B.2-76	87
FII.B.2-77	87
FII.B.2-78	87
FII.B.2-79	87
FII.B.2-80	87
FII.B.2-81	93
FII.B.2-82	93
FII.B.2-83	93
FII.B.2-84	93
FII.B.2-85	94
FII.B.2-86	94
FII.B.3-1	100
FII.B.3-2	52
Section II.D	104
TII.D.1-1	104
Section II.E	104
TII.E.1.1-1	104
TII.E.1.1-2	104

(continued)

Page	Amendment No.	Page	Amendment No.
TII.E.1.1-3	104		
TII.E.1.1-4	104		
TII.E.1.1-5	Deleted		
FII.E.1.1-1	104		
FII.E.1.1-2 (Sh. 1)	January 30, 1981		
FII.E.1.1-2 (Sh. 2)	January 30, 1981		
FII.E.1.1-2 (Sh. 3)	January 30, 1981		
FII.E.1.1-2 (Sh. 4)	January 30, 1981		
FII.E.1.1-2 (Sh. 5)	January 30, 1981		
Notes to FII.E.1.1-2	January 30, 1981		
Section II.F	105		
TII.F.2-1	104		
FII.F-1	December 10, 1982		
Section II.G	104		
Section II.K	104		
Section III.A	104		
FIII.A.1.2-1	95		
FIII.A.1.2-2	83		
FIII.A.1.2-3	95		
FIII.A.1.2-4	95		
Section III.D	104		
EL-1 thru EL-55	105		

(continued)

Changes incorporated by approved LDCRs since issuance of Amendment 100

Amendment No.	LDCR No./Licensing Lead	Effective Sections
100, Supplement a	SA-2006-12 (RAS)	All Sections, Tables, and Figures
100, Supplement b	SA-2005-8 (DWS)	T17A-1
	SA-2005-22 (DWS)	17.2, 1A(B)
	SA-2005-33 (DWS)	17.2
	SA-2006-11 (DWS)	17.2
	SA-2006-16 (DWS)	17.2
	SA-2006-9 (DWS)	Chap. 17 TOC, 17.2, Delete F17.2-1, Delete F17.2-2
	SA-2006-14 (DWS)	17.2
	SA-2006-22 (DWS)	17.2
	SA-2005-15 (RJK)	T6.3-2, 3.6B, T6.3-1
	SA-2004-47 (MJR)	T6.2.4-1
	SA-2006-20 (DWS)	13.4, 13.3B
	SA-2006-7 (DWS)	13.1, T13.1-1, 13.1A, F13.1-2, F13.1-3, 13.3B, TMI-I.C, 13.1A
	SA-2006-17 (JCH)	T10.4-11
	SA-2003-47 (MJR)	9.5, 1A(B), 8.3, 9.2, T8.3-3, 9.5
	SA-2005-32 (TJE)	1A(B), 8.3, T8.3-10 - sheet 9 & 10
	SA-2005-2 (TJE)	8.3

Changes incorporated by approved LDCRs since issuance of Amendment 100

Amendment No.	LDCR No./Licensing Lead	Effective Sections
100, Supplement b (continued)	SA-2005-7 (TJE)	8.3
	SA-2005-19 (TJE)	8.2
	SA-2005-25 (TJE)	1A(B), 8.3
	SA-2006-6 (TJE)	1.6, T1.6-1 - sheet 2 & 3
	SA-2004-43 (CBC)	F3.6B-64-1, F3.6B-64-2
	SA-2005-31 (CBC)	8.2
	SA-2005-23 (CBC)	T3.9B-10
	SA-2006-21 (DWS)	13.1A
	SA-2004-19 (CBC)	3.3, 9.4C, T9.4-2
	SA-2005-20 (CBC)	9.1
	SA-2005-18 (CBC)	9.1
	SA-2005-28 (CBC)	9.1
	SA-2004-36 (CBC)	1.2, 6.4, T6.4-4, 9.4, T9.4-10, 15.6, F9.4-1
	SA-2005-10 (TJE)	F8.3-16
	SA-2004-22 (GLM)	9.2
	SA-2004-44 (GLM)	9.3, T9.3-1, 8.3, T8.3-10, T8.3-11
	SA-2005-27 (GLM)	9.2

Amendment No.	LDCR No./Licensing Lead	Effective Sections
100, Supplement b (continued)	SA-2006-8 (GLM)	T9.4-2
	SA-2006-19 (GLM)	9.3, 10.4
101 (February 1, 2007)	SA-2005-13 (JCH)	10.4
	SA-2005-30 (JCH)	10.4
	SA-2006-26 (GLM)	10.4
	SA-2005-17 (JDS)	T7.5-7E - sheet 4 of 4
	SA-2005-24 (JDS)	1A(B), 6.2
	SA-2006-13 (JDS)	4.4, 5.1
	SA-2005-26 (JDS)	F4A.2.2-1, T4A.2.2-2, 4.3, 4A, T4A1.2-1, T4A.2.2-1, 4.4, 4.2, T1.6-1
	SA-2006-32 (RJK)	T9.3-4
	SA-2006-15 (TJE)	8.3
	SA-2006-33 (TJE)	8.3
	SA-2006-5 (RJK)	3.6B, T6.3-1, T6.3-2
	SA-2006-18 (NSH)	T10.4-18
	SA-2006-27 (JCH)	13.3
	SA-2006-29 (DWS)	T17A-1
	SA-2006-28 (JDS)	T4B.2.2-2, F4B.2.2-1, F4B.2.2-2, F4B.2.2-3, F4B.2.2-4, 15.6, 4B, T4B.1.2-1, T4B.2.2-1, 9.1.4

Amendment No.	LDCR No./Licensing Lead	Effective Sections
101 (February 1, 2007) (continued)	SA-2007-1 (DWS)	14.2, F13.1-2, F13.1-3, 12.1, I.B, I.C, CHAP. 13 TOC, 13.1, T13.1, 13.1A, 13.3, 13.4, 13.5, CHAP. 17 TOC, 17.2, 17A
	SA-2004-42 (JCH)	3.5, 5.4, 7.1, 7.3, T7.3-4, 7.7, 9.5, CHAP. 10 TOC, 10.2
	SA-2005-16 (JCH)	3.9B
101a August 09, 2007	SA-2006-2 (TJE)	1A(B), 3.7B, F3.7B-54
	SA-2007-4 (TJE)	1A(B), 9.5
	SA-2007-3 (GLM)	9.3
	SA-2006-4 (RJK)	1.2, F1.2-1
	SA-2005-12 (CBC)	1A(B), 6.4, T6.4-1, T6.4-3, T6.4-4, 6.5, T6.5-5, T6.5-6, T6.5-7, Ch. 15 LOT, T15.0-7, Delete T15.6-8, 15.1, T15.1-3, T15.1-4, Delete T15.1-4A, 15.3, T15.3-1, 15.4, T15.4-4, 15.6, T15.6-2, T15.6-3, T15.6-4, T15.6-9, T15.6-10, 15.7, T15.7-1, T15.7-2, T15.7-3, T15.7-4, T15.7-6, T15.7-7, Appendix 15B, T15B-1
	SA-2007-12 (TJE)	8.2
	SA-2006-40 (RJK)	Ch. 1 TOC, 1.2, T1.6-2, 1A(N), T1A(N)-1, Ch. 3 LOT, 3.6B, T3.6B-4A, T3.6B-4B, T3.6B-6, 3.7N, 3.8, 3.9N, F3.9N-1, T3.9N-14A, T3.9N-14B, T3.9N-15A, T3.9N-15B, T3.9N-16A, T3.9N-16B, T3.9N-19A, T3.9N-19B, T3.9N-21A, T3.9N-21B, F4.4-21A, F4.4-21B, Ch. 5 TOC, Ch. 5 LOT, T5.1-1A, T5.1-1B, 5.2, T5.2-1, T5.2-2, T5.2-6A, T5.2-6B, T5.2-7, 5.3, T5.3-16A, T5.3-17A, 5.4, T5.4-3, T5.4-4A, T5.4-4B, F5.4-4A, F5.4-4B, F5.4-13, 5A, 12.1, 12.3
	SA-2006-42 (GLM)	F1.2-1, 3.8
	SA-2006-37 (JCH)	10.3, 10.4

Amendment No.	LDCR No./Licensing Lead	Effective Sections
101a August 09, 2007 (continued)	SA-2006-38 (JCH)	3B, 3.6B, 3.9B, 7.1, 7.3, 10.4, T3.6B-1, T3.9B-1F, T3.9B-10, T6.2.4-2, T6.2.4-3, T6.2.4-6, T7.3-4, T7.5-5, T7.5-7B, T10.3-6, F3.6B-11, F3.6B-12, F3.6B-13, F3.6B-14, F3.6B-19, F3.6B-20, F3.6B-21, F3.6B-22, F3.6B-26, F3.6B-27, F3.6B-28, F3.6B-29, F3.6B-30, F3.6B-34, F3.6B-35, F3.6B-36, F3.6B-37, F3.6B-38, F3.6B-39-1, F3.6B-40, F3.6B-41, F3.6B-55, F3.6B-67-1, F3.6B-82-1, F3.6B-208 sh. 2, F3.6B-208 sh. 3, F6.2.4-1 sh. 9, F7.7-14, F10.4-12, F10.4-22A, F10.4-22B, FII.E.1.1-1
	SA-2006-39 (JDS)	1A(B), Ch. 3 LOT, Ch. 3 LOF, 3.5, T3.5-2A, T3.5-2B, T3.5-6, 3.7N, 3.8, T3.8-1, 3.9N, T3.9N-20, 4.5, 5,2, 5.3, T5.3-2A, T5.3-5, T5.3-15A, T5.3-1, 6.2, 7.7, 8A, 8.3, 9.1, T9.4-2, 12.3, T17A-1, II.F, F3.9N-5A, F3.9N-5B, F3.9N-6A, F3.9N-6B, F3.5-1, F3.5-2, F1.2-8, F1.2-13, F3.8-15, F3.8-2, F5.1-3
	SA-2007-6 (RJK)	T1.6-1, 4A, 15.6
	SA-2003-21 (JCH)	T3.9B-10
	SA-2007-9 (RJK)	6.3
101b February 4, 2008	SA-2006-1 (RJK)	12.3, F12.3-33 sh. 3, F12.3-6, F12.3-23.4
	SA-2007-17 (RJK)	T1.6-1, 5.2, Ch. 5 LOF, Deleted F5.2.2, Deleted F5.2.3
	SA-2004-30 (TJE)	1.2, 1A(N), 1A(B), 3.1, T3.10N-1, Ch. 6 TOC, Ch. 6 LOT, Ch. 6 LOF, 6.2, T6.2.5-1, T6.2.5-2, T6.2.5-4, T6.2.5-5, T6.2.5A-1, T6.2.5A-2, T6.2.5A-3, T6.2.5A-5, T6.2.5A-6, F6.2.5-1, Delete F6.2.5-2, Delete F6.2.5A-1, Delete F6.2.5A-2, Delete F6.2.5A-3, Delete F6.2.5A-5, Delete F6.2.5A-7, Delete F6.2.5A-9, 7.1, T7.1-2.6, 7.2, 7.3, T7.5-3, T7.5-4, T7.5-7A, T7.5-7D, T7.5-7E, 7.6, T8.1-1, T8.3-1, T8.3-2, T13.5-2, 15.6, T17A-1, TII.B.2-4, TII.B.2-5, TII.B.2-6, II.E, II.F

Amendment No.	LDCR No./Licensing Lead	Effective Sections
101b February 4, 2008 (continued)	SA-2007-11 (JDS)	CH. 6 LOT, CH. 6 LOF, 6.2, T6.2.1-2A.1, T6.2.1-2A.2, Delete T6.2.1-2A, T6.2.1-2B.1, T6.2.1-2B.2, Delete T6.2.1-3, Delete T6.2.1-3A, Delete T6.2.1-4, Delete T6.2.1-4A, T6.2.1-5, Delete T6.2.1-9, T6.2.1-9A, T6.2.1-9B, Delete T6.2.1-10, T6.2.1-10A, T6.2.1-10B, Delete T6.2.1-15, Delete T6.2.1-21, Delete T6.2.1-26, Delete T6.2.1-29, Delete T6.2.1-30, Delete T6.2.1-37, Delete T6.2.1-38, Delete T6.2.1-39, Delete T6.2.1-40, Delete T6.2.1-41, Delete T6.2.1-42, Delete T6.2.1-43, Delete T6.2.1-44, Delete T6.2.1-45, Delete T6.2.1-46, Delete T6.2.1-47, Delete T6.2.1-48, Delete T6.2.1-50, Delete T6.2.1-50A, Delete T6.2.1-51, Delete T6.2.1-52, Delete T6.2.1-53, Delete T6.2.1-54, Delete T6.2.1-55, Delete T6.2.1-56, Delete T6.2.1-57, Delete T6.2.1-58, Delete T6.2.1-59, Delete T6.2.1-60, Delete T6.2.1-64, Delete T6.2.1-65, Delete T6.2.1-66, Delete T6.2.1-67, Delete T6.2.1-68, Delete T6.2.1-69, Delete T6.2.1-77, Delete T6.2.1-91, Delete T6.2.1-93, Delete T6.2.1-95, Delete T6.2.1-96, Delete T6.2.1-97, Delete T6.2.1-98, Delete T6.2.1-99, Delete T6.2.1-100, Delete T6.2.1-101, Delete T6.2.1-102, Delete T6.2.1-103, Delete T6.2.1-104, Delete T6.2.1-1, Delete T6.2.1-2, Delete T6.2.1-3, F6.2.1-9, F6.2.1-10, F6.2.1-11, F6.2.1-12, F6.2.1-13, F6.2.1-14, F6.2.1-15, F6.2.1-16, Delete F6.2.1-20, Delete F6.2.1-, Delete F6.2.1-30, Delete F6.2.1-31, Delete F6.2.1-32, Delete F6.2.1-33, Delete F6.2.1-34, Delete F6.2.1-35, Delete F6.2.1-36, Delete F6.2.1-37, Delete F6.2.1-38, Delete F6.2.1-39, Delete F6.2.1-40, Delete F6.2.1-41, Delete F6.2.1-2, Delete F6.2.1-43, Delete F6.2.1-44, Delete F6.2.1-45, Delete F6.2.1-49, Delete F6.2.1-50, Delete F6.2.1-51, Delete F6.2.1-52, Delete F6.2.1-54, Delete F6.2.1-55, Delete F6.2.1-56, Delete F6.2.1-58, Delete F6.2.1-59, Delete F6.2.1-60, Delete F6.2.1-61, Delete F6.2.1-62, Delete F6.2.1-63, Delete F6.2.1-64, Delete F6.2.1-65, Delete F6.2.1-66, Delete F6.2.1-67, Delete F6.2.1-68, Delete F6.2.1-69, Delete F6.2.1-70

Amendment No.	LDCR No./Licensing Lead	Effective Sections
101b February 4, 2008 (continued)	SA-2006-10 (JDS)	1A(B), T3.9B-10, 6.2, F6.2.2-3, F6.2.2-3A, F6.2.2-4, F6.2.2-5, 6.3, 6.5, T6.5-4, T6.3-7, T6.3-11, 7.6, T17A
	SA-2006-31 (JDS)	9.1
	SA-2007-19 (JDS)	6.2
	SA-2007-23 (GLM)	9.1, T1.6-1
	SA-2007-22 (JDS)	1A(B), 6.1B, 6.2, T17A-1
	SA-2006-24 (JCH)	T17A-1, T3.9B-10
	SA-2006-34 (JCH)	10.4
	SA-2007-16 (RJK)	5.3
	SA-2007-5 (JDS)	F4A.2.2-1, F4A.2.2-2, F4A.2.2-3, 4A, T4A.1.2-1, T4A.2.2-1, T4A.2.2-2, 15.0, 15.1, 15.2, 15.4, 15.5, 15,6, T15.6-5
	SA-2007-21 (GLM)	9.2
102 August 1, 2008	SA-2007-18 (RJK)	5.3
	SA-2007-20 (TJEW)	9.1, T9.1-1, T12.2-24, 15.7, T15.7-6, T15.7-7
	SA-2006-41 (CBC)	6.2
	SA-2006-23 (JCH)	9.5
	SA-2005-29 (JDS)	T7.5-7B, F5.4-6, F6.2.2-1
	SA-2008-4 (RJK)	9.3, T9.3-6, T9.3-7, 15.6
	SA-2006-3 (JCH)	10.2

Amendment No.	LDCR No./Licensing Lead	Effective Sections
102 August 1, 2008 (continued)	SA-2007-2 (TJEW)	T1.6-1, 3.5, 10.2,
	SA-2006-25 (JCH)	10.2, T10.4-10, T10.1-1, T10.4-17
	SA-2008-8 (RAS)	Ch. 13 TOC, Ch. 13 LOF, 13.1, T13.1-1, 13.1A-1, 13.3, 13.5, F13.1-1, F13.1-2, F13.1-3, Ch. 17 TOC, 17.1, 17.2
	SA-2008-9 (RJK)	5.2
102a February 2, 2009	SA-2008-14 (TJEW)	13.1
	SA-2008-15 (JDS)	12.3
	SA-2007-15 (JCH)	10.4
	SA-2006-36 (JDS)	6.2, T6.1B-3, T6.2.2-4, 6.3,
	SA-2008-17 (RAS)	6.4
	SA-2008-16 (JCH)	10.2
	SA-2008-1 (JDS)	6.2
102b August 3, 2009	SA-2008-2 (GLM)	9.4E
	SA-2009-13 (GLM)	I.D-1
	SA-2009-8 (JCH)	10.2, F10.2-1
	SA-2008-22 (SCD)	12.5
	SA-2009-2 (TJD)	17.2
	SA-2009-6 (JDS)	5.4, 6.3

Amendment No.	LDCR No./Licensing Lead	Effective Sections
102b August 3, 2009 (continued)	SA-2008-10 (JCH)	5.3, T5.3-4C
	SA-2009-11 (JDS)	T5.2-2,1A(N)
	SA-2008-23 (TJD)	17.2, Ch. 17 TOC
103 February 11, 2010	SA-2004-26 (SCD)	T10.4-10
	SA-2009-5 (CBC)	I.A
	SA-2009-7 (TJD)	17.2
	SA-2009-17 (TJD)	1A(B), 17.2
	SA-2009-3 (SCD)	Ch. 13 TOC, 13.1, T13.1-1, 13.1A, 13.3, 13.5
	SA-2009-10 (TJD)	T2.5.6-7, F2.5.6-56
	SA-2009-16 (TJEW)	T8.3-3
	SA-2009-23 (TJEW)	T8.3-4
	SA-2008-11 (JCH)	10.2, 10.4, T10.4-10
	SA-2008-12 (JCH)	10.4
	SA-2008-20 (JCH)	13.3
	SA-2009-19 (JDS)	4.2, 9.3
	SA-2007-10 (JDS)	1.2, T10.1-1, Ch. 10 LOF, 10,1, 10.2, 10.4, F10.1-1, F10.1-2
	SA-2004-18 (JDS)	5.3

Amendment No.	LDCR No./Licensing Lead	Effective Sections
103 February 11, 2010 (continued)	SA-2008-13 (JDS)	T1.6-1, Ch. 6 TOC, Ch. 6 LOT, Ch. 6 LOF, 6.2, T6.2.1-2, T6.2.1-2A, T6.2.1.3, T6.2.1-3A, T6.2.1-3B, T6.2.1-4, T6.2.1-5, T6.2.1-6, T6.2.1-8, T6.2.1-9, T6.2.1-10, T6.2.1-11, T6.2.1-12, T6.2.1-39, T6.2.1-41, T6.2.1-42, T6.2.1-43, T6.2.1-49, T6.2.1-50, T6.2.1-51, T6.2.1-54, T6.2.1-56, T6.2.1-57, F6.2.1-1, F6.2.1-2, F6.2.1-3, F6.2.1-4, Ch. 7 TOC, 7.7, 9.3, Ch. 15 LOT, Ch. 15 LOF, 15.0, T15.0-1, T15.0-4, T15.0-6, T15.0-7, F15.0-1, F15.0-3, F15.0-4, F15.0-5, 15.1, T15.1-3, T15.1-4, 15.2, T15.3-1, 15.4, T15.4-4, 15.5, 15.6, T15.6-1, T15.6-2, T15.6-3, T15.6-20, T15.6-21, F15.6-1, F15.6-2, F15.6-3, F15.6-4, F15.6-5, F15.6-6, F15.6-7, F15.6-8, F15.6-9, F15.6-10, F15.6-11, F15.6-12, F15.6-13, F15.6-14, F15.6-15, F15.6-16, F15.6-17, F15.6-18, F15.6-19, F15.6-20, F15.6-21, F15.6-22, F15.6-23, F15.6-24, F15.6-25, F15.6-26, F15.6-27, F15.6-28, F15.6-29, F15.6-30, F15.6-31, F15.6-32, F15.6-33, F15.6-34, F15.6-35, F15.6-36, F15.6-37, F15.6-38, F15.6-39, F15.6-40, F15.6-41, F15.6-42, F15.6-43, F15.6-44, F15.6-45, F15.6-46, F15.6-47, F15.6-48, F15.6-49, F15.6-50, F15.6-51, F15.6-52, F15.6-53, F15.6-54, F15.6-55, F15.6-56, F15.6-57, F15.6-58, F15.6-59, F15.6-60, F15.6-61, F15.6-62, F15.6-63, F15.6-64, F15.6-65, F15.6-66, F15.6-67, F15.6-68, F15.6-69, F15.6-70, F15.6-71, F15.6-72, F15.6-73, F15.6-74, F15.6-75, F15.6-76, F15.6-77, F15.6-78, F15.6-79, F15.6-80, 15.7, T15.7-1, T15.7-2, T15.7-3, T15.7-4, T15.7-6, T15.7-7
103a (August 2, 2010)	SA-2008-19 (JCH)	13.3
	SA-2010-2 (JCH)	13.3
	SA-2009-24 (JDS)	6.2
	SA-2009-12 (SCD)	Ch. 6 TOC, 5.2, 6.2, 6.3, F6.2.4-1, Shts. 3 & 12

103a (August 2, 2010) (continued)	SA-2010-11 (TJD)	1A(B), II.K
	SA-2009-25 (TJEW)	T3.2-3, T3.2-4, Ch. 9 LOF, Deleted M1-0313A (F9.4-15)
	SA-2009-21 (JDS)	Ch. 15 TOC, 15.4
	SA-2010-10 (JDS)	7.2, T7.2-4, T15.0-6, 15.4
103b (February 1, 2011	SA-2010-6 (JCH)	T10.4-13
	SA-2009-9 (RAS)	3.8, 9.1
	SA-1010-7 (SCD)	11.3
	SA-2010-13 (TJD)	2.4
	SA-2010-9 (TJD)	Ch. 17 TOC, 17.2
	SA-2011-1 (TJEW)	T5.1-1B
104 (August 1, 2011)	SA-2010-18 (RAS)	7.4, T7.4-1, T7.4-3
	SA-2009-14 (SCD)	6.3
	SA-2011-8 (SCD)	11.4, F1.2-1
	SA-2010-20 (TJD	T17A-1
	SA-2008-21 (JCH)	T17A-1
	SA-2010-5 (JDS)	4.2, 1A(N)
	SA-2010-8 (JDS)	7.6, CH. 9 LOF, 9.1, F9.1-4, 9.5, T17A-1,
	SA-2011-5 (JDS)	17.2, 17A
	SA-2010-16 (JCH)	9.5

104 (August 1, 2011) (continued)	SA-2011-3 (SCD)	12.1, 12.5, T13.1, 13.1A
	SA-2010-17 (TJEW)	3.11B
	SA-2009-4 (TJEW)	F1.2-1, Ch. 8 LOF, 8.2, F8.2-1, F8.2-4, F8.2-5 Sh. 1, F8.2-10 Sh. 1 & 2, F8.2-11, 8.3, 9.5, F10.2-1
	SA-2011-6 (JDS)	1.2, F1.2-1, Ch. 9 TOC, 9.1, 9.5, 12.4
	SA-2011-12 (TJEW)	9.5, F9.5-51
	SA-2010-19 (TJEW)	10.4
	SA-2009-20 (TJEW)	1.2, 8.1, 8.2, 8.3, F8.2-1, F8.2-4, F8.2-5 Sht. 1, F8.2-5 Sht. 2, F8.2-12
	SA-2010-1 (SCD)	11.1, T11.1-1, T11.1-2, T11.1-3, T11.1-4, 11.1-5, T11.1-7, 11.2, T11.2-1, T11.2-2, T11.2-8, T11.2-9, T11.2-10, 11.2, T11.3-3, T11.3-4, T11.3-5, T11.3-7, 11A, T11A-1, T11A-2, T11A-3, T11A-4, T11A-5, T11A-6, 121.2, T12.2-1, T12.2-2, T12.2-3, T12.2-4, T12.2-5, T12.2-7, T12.2-8, T12.2-9, T12.2-10, T12.2-11, T12.2-12, T12.2-13, T12.2-14, T12.2-15, T12.2-17, T12.2-17A, T12.2-19, T12.2-19A, T12.2-19B, T12.2-19C, T12.2-19D, T12.2-19E, T12.2-21, T12.2-22, T12.2-23, T12.2-24, T12.2-25, T12.2-26, 6.2, 3.6B, T3.6B-4A, T3.6B-4B, 9.1, 9.2, 10.4, T10.4-10
104a (February 1, 2012)	SA-2010-14 (GLM)	9.2
	SA-2011-13 (JDS)	T1.6-1, 4.1, T4.1-1, 4.2, 4.4, T4.4-1, T15.6-4, T15.6-5
	SA-2011-14 (JDS)	15.5
	SA-2011-2 (RAS)	7.2
	SA-2009-22 (TJD)	17.2
	SA-2011-15 (TJEW)	T3A.3-2

104a (February 1, 2012) (continued)	SA-2011-18 (TJEW)	9.5
	SA-2009-18 (TJEW)	7.5, T7.5-2, T7.5-7A, T7.5-7C, T7.5-7D, T7.5-7E, II.F-3
	SA-2011-21 (TJEW)	T7.5-7B
	SA-2011-17 (JDS)	T3A.3-1, 6.2
	SA-2011-7 (JDS)	T3.7B(A)-1
	SA-2011-16 (SCD)	12.1, Ch. 13 TOC, 13.1, 13.1A, 13.4, 14.2, Ch. 17 TOC, 17.2
104b (August 1, 2012)	SA-2012-9 (RAS)	7.7
	SA-2012-7 (TJEW)	T3.2-3, T3.2-4, Ch. 6 LOF
	SA-2012-11 (JCH)	T3.9N-10
	SA-2012-10 (SCD)	Ch. 11 TOC, 11.4, Ch. 12 TOC, 12.2
	SA-2012-14 (SCD)	T17A-1
	SA-2012-17 (SCD)	3.4
	SA-2010-3 (TJEW)	1A(B), 9.5
	SA-2012-15 (SCD)	1A(B), T6.5-1
105 (February 4, 2013)	SA-2011-11 (GLM)	T9.4-2, 9.4C
	SA-2011-19 (GLM)	9.2
	SA-2012-2 (TJEW)	T7.5-7B
	SA-2012-28 (JDS)	15.5
	SA-2012-23 (CBC)	1A(B)

105 (February 4, 2013) (continued)	SA-2010-12 (GLM)	9.4E
	SA-2012-22 (JCH)	3.6B
	SA-2012-6 (JDS)	9.1
	SA-2012-16 (SCD)	11.5, F11.5-1
	SA-2012-26 (TJEW)	T3.9B-10
	SA-2012-4 (TJEW)	9.5, T17A-1
	SA-2012-8 (JDS)	3.5, T7.5-7B, 10.4, 15.1, T17A-1
	SA-2012-19 (TJEW)	T3A.3-2, 5.3, 6.2, 8.3
	SA-2012-24 (JDS)	15.6, T15.6-8, T15.6-9, T15.6-10, T15.6-11
	SA-2013-2 (TJEW)	T3.9B-10
	SA-2012-18 (JDS)	Ch. 6 LOF, 6.2, T6.2.1-2, Deleted F6.2.1-1 and F6.2.1-2
	SA-2012-27 (SCD)	12.5
	SA-2012-21 (SCD)	13.1, T13.1-1, 13.1A
	SA-2012-20 (SCD)	17.2
	SA-2013-3 (SCD)	F9.2-4A, F9.3-1
105a (August 5, 2013)	SA-2013-5 (TJEW)	F8.2-4, F8.2-5, Sh. 1
	SA-2013-1 (JDS)	4.1, 4.2
	SA-2013-8 (JDS)	T6.3-11
	SA-2012-12 (JDS)	T3.9B-10

105a (August 5, 2013) (continued)	SA-2012-5 (JCH)	10.4
	SA-2012-1 (JCH)	5A, T9.3-9
105b (February 5, 2014)	SA-2013-10 (JCH)	10.2
	SA-2013-9 (JDS)	9.3
	SA-2013-20 (JDS)	15.4
	SA-2013-22 (RAS)	7.7
	SA-2013-23 (RAS)	F7.3-4
	SA-2013-12 (TJEW)	8B-2
	SA-2013-19 (RAS)	3.6B
	SA-2013-24 (SCD)	Ch. 17 TOC, 17.2, 17A, TMI I.B
	SA-2013-15 (GLM)	9.2
	SA-2013-4 (GLM)	3.5
	SA-2013-17 (JDS)	4.3, 6.2
	SA-2013-21 (GLM)	9.2
106 July 31, 2014	SA-2013-7 (GLM)	9.2
	SA-2013-26 (GLM)	2.4, 9.2
	SA-2014-1 (JDS)	T15.6-8, T15.6-10
	SA-2014-2 (SCD)	6.3
	SA-2014-3 (SCD)	T17A-1

106 July 31, 2014 (continued)	SA-2014-5 (GLM)	1.2
	SA-2014-7 (GLM)	Ch. 9 LOF, F9.3-1
	SA-2014-6 (SCD)	T12.5-1
	SA-2014-9 (SCD)	6.3, T6.3-1
	SA-2014-11 (TJEW)	8.3
	SA-2012-13 (TJEW)	F1.2-1, 8.2, F8.2-1, F8.2-4, F8.2-7 Sh. 1 & 2, F8.2-9, F8.2-11, F8.2-11A, 9.5
	SA-2014-15 (TJEW)	8.3
	SA-2014-14 (SCD)	Ch. 17 TOC, 17.2, TMI I.B
	SA-2014-16 (JDS)	1.1
	SA-2014-17 (JDS)	9.1
	SA-2014-18 (TJEW)	Ch. 8 LOF
	SA-2014-13 (SCD)	Ch. 13 TOC, 13.1, T13.1-1, F13.1-2, F13.1-3, 13.1A
	SA-2011-4 (JDS)	15.6
106a February 2, 2015	SA-2014-20 (JCH)	T3.9B-10
	SA-2013-6 (JDS)	T4.1-2, 4.3, 9.1 T9.1-4
	SA-2014-21 (SCD)	T9.3-7
	SA-2014-26 (JDS)	T6.2.4-3
	SA-2014-12 (SCD)	T17A-1
	SA-2014-22 (SCD)	T3.2-3, T3.2-4, Ch. 11 LOF, F11.3-1

106a February 2, 2015 (continued)	SA-2014-19 (SCD)	6.2, T17A-1
	SA-2014-25 (JDS)	T15.0-4
	SA-2015-1 (SCD)	9.3, T17A-1
106b September 3, 2015	SA-2015-4 (JEB)	13.6
	SA-2014-23 (RAS)	10.4
	SA-2015-7 (SCD)	T6.5-1
	SA-2014-28 (JDS)	T3.7BA-1
	SA-2015-8 (JDS)	9.1
	SA-2015-10 (JDS)	1A(B), 6.2, 6.3, T6.3-7
107 February 1, 2016	SA-2015-12 (SCD)	6.3
	SA-2015-13 (GLM)	9.2
	SA-2015-15 (GLM)	9.2
	SA-2015-18 (SCD)	T6.3-3
	SA-2015-17 (GLM)	Chapter 9 LOF
	SA-2014-4 (CBC)	1.2, F1.2-1, 9.1, T17A-1, T17A-2
	SA-2015-16 (SCD)	17.2
	SA-2014-27 (JDS)	6.2, T6.2.1-2, T6.2.1-2a
	SA-2015-19 (SCD)	Chapter 13 TOC, 13.1, T13.1-1, 13.1A, F13.1-2

FSAR Amendment 100, Supplement a

LDCR-SA-2006-12 (RAS):

Administrative change for software conversion only. The changes are editorial in nature and contain no technical changes. The electronic files have been converted from Microsoft WORD to Adobe FrameMaker and published in Adobe Portable Document Format (pdf). The type of changes include changes such (1) correction of spelling errors, (2) correction of inadvertent word processing errors from previous changes, and (3) style guide changes (e.g., changing from a numbered bullet list to an alphabetized bullet list and vice versa, change numbering of footnote naming scheme). The entire FSAR will be reissued as Amendment 100, Supplement a except as noted below. For the text and tables there will be no change bars in the page margins for the editorial changes. The list of effective pages is being replaced with a list of effective sections, tables, and figures. Some Figures and Tables will retain an Amendment number prior to Amendment 100 since the source file is not in Adobe FrameMaker (e.g., scanned page).

Sections Revised: All

Tables Revised: All (except as noted above)

Figures Revised: All (except as noted above)

FSAR Amendment 100, Supplement b

LDCR-SA-2005-8, EVAL-2003-002423-02 (DWS):

Table 17A-1, sheets 45, 53 & 58

Revise the Q-List criteria to add instrument tubing that is connected to ANSI Safety Class 3 ductwork to clarify that criteria appropriate for the application is not the same as for ASME applications covered by FSAR Section 3.9B.3.4.3.

Although not included in commitment CDF-22112, FSAR Table 17A-1 previously applied the same criteria for instruments connected to Safety Class 3 HVAC ductwork as for ASME piping. The change to Table 17A-1 would document that this very conservative practice may be continued; however, it is not required. The tubing should only have to meet the seismic requirements comparable to the equipment to which it is connected.

Section 9.4: FSAR Figure 9.4-9 is revised to show the impulse lines to the non-safety instruments on the Non-ESF PPVES units are being declassified to NNS. This would allow seismic Category II supports rather than seismic Category I supports.

Where the tubing is reclassified as NNS in accordance with ANSI N18.2 [as described in DBD-ME-028], the instrument should only have to meet the requirements of the tubing. FSAR Figure 9.4-9 is revised to show the impulse lines to the non-safety instruments on the Non-ESF PPVES units are being declassified to NNS based on no credible failure which could adversely affect the Nuclear Safety Function of the PPVES.

LDCR-SA-2005-22, EVAL-2005-002874-01 (DWS):

Section 17.2.12

Revise paragraph on M&TE and reference standards to indicate the standards are traceable to nationally and internationally recognized standards. National laboratories no longer support all types of calibration.

NUREG-0800 Standard Review Plan Section 7.1 "Instrumentation and Controls - Introduction" endorses Branch Technical Position HICB-12 "Guidance on Establishing and Maintaining Instrument Setpoints" which endorses ANSI/NCSL Z540-1-1994 "American National Standard for Calibration _ Calibration Laboratories and Measuring and Test Equipment _ General Requirements".

ANSI/NCSL Z540-1-1994 Section 9.2 states in part ".....measurements made by the laboratory are traceable to national, international, or intrinsic standards....

LDCR-SA-2005-22, EVAL-2005-002874-01 (DWS) (continued):

Section 1A(B) - Regulatory Guide 1.30

NUREG-0800 Standard Review Plan Section 7.1 "Instrumentation and Controls - Introduction" endorses Branch Technical Position HICB-12 "Guidance on Establishing and Maintaining Instrument Setpoints" which endorses ANSI/NCSL Z540-1-1994 "American National Standard for Calibration _ Calibration Laboratories and Measuring and Test Equipment _ General Requirements".

ANSI/NCSL Z540-1-1994 Section 9.2 states in part ".....measurements made by the laboratory are traceable to national, international, or intrinsic standards...."

LDCR-SA-2005-33, EVAL-2004-003442-01 (DWS):

Delete from Section FSAR 17.2.1.3.2 and 17.2.18.2 as follows "For audits other that those audit areas with a maximum frequency specified by regulation (Security, Emergency Planning, (Delete "and Radiation Protection") a grace period..........

Historically the periodic review described in 10CFR20.1101(c) has been satisfied by an independent quality assurance audit. This annual review will be satisfied as described below and every 24 months a quality assurance audit will be performed which is consistent with ANSI N18.7 sect 4.5

NOD Audits (which usually include industry peers)

* NOD currently performs annual audits.

NOD Surveillances

Manager and Supervisory Observations (LEADING)

Annual performance reviews

Corrective Action Program (documentation of procedural noncompliance in SmartForms)

RP Self- Assessments (which may include industry peers or contractors)

Third Party Audits (NRC, INPO, NVLAP and ANI)

FSAR Amendment 100, Supplement b

LDCR-SA-2006-11, EVAL-2006-000798-01 (DWS):

Section 17.2: Revise ORC membership composure, definition of a ORC quorum, and audit functions. Move audits listed under Section 17.2.1.3.2 to Section 17.2.18

This is an administrative change only and is in compliance with the requirements of ANSI N18.7-1976.

LDCR-SA-2006-16, EVAL-2006-001294-01 (DWS):

Section 17.2: Add statement "TXU Power may delegate to others such as contractors, agents, or consultants the work of establishing and executing the quality assurance program, or any part thereof". And clarify QA responsibilities on page 21, 23, 28 and 29.

Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," allows the use of contractors to perform QA functions as long as the overall responsibility for the Quality Assurance (QA) Program remains with the Senior Vice President & Chief Nuclear Officer.

LDCR-SA-2006-9, EVAL-2005-003061-02 (DWS):

Section 17.2: Revise composure of the Station Operations Review Committee (SORC) and current organizational positions. Replace "Vice President, Nuclear Engineering" with "Vice President, Nuclear Engineering and Support" wherever it appears. Replace "Vice President, Nuclear Operations" with "Site Vice President" wherever it appears. Define the responsibilities of the Site Vice President and Plant Manager. Delete some references to Section 13.1.1.2.1. Insert Figure 13.1-2 and 13.1-3 for Figures 17.2-1 and 17.2-2.

Revised composure of SORC will reflect current organizational changes. This would resolve the discrepancy between current directors and the generic title of managers, clearly indicate the minimum number of SORC members and clarify that the listed individuals are in areas of experience, not organizational or position titles.

LDCR-SA-2006-14, EVAL-2006-001365-01 (DWS):

Section 17.2.1.3.1: Revise FSAR Section 17.2.1.3.1 to delete ORC review of changes to the Technical Specifications or the Operating License. Move this review requirement to Quality Assurance FSAR Section 17.2.1.6

American National Standard N18.7-1976/ANS-3.2, Section 4.3 requires the independent review of operational activities and allows these activities to be performed either by a Standing Committee (ORC) (Section 4.3.2) or by Organizational Units (Section 4.3.3). This change deletes the use of ORC for review of changes to the Technical Specifications or Operating License and incorporates the alternative method of using Organizational Units to perform the Independent Review.

FSAR Amendment 100, Supplement b

LDCR-SA-2006-22, EVAL-2006-002136-01 (DWS):

17.2.1.1.3.1 change reference for required experience from ANSI N18.1-1971 to Regulatory Guide 1.8, Rev.2. The RG is a higher tier document, therefore there is no change in commitment.

17.2.1.3 change reference to items "a through h" above to "1 through 9". This is editorial only to match the new format.

17.2.3 remove requirement for ORC approval before proceeding with implementation of TS changes or other changes to the operating license. This change was approved in LDCR SA-2006-14, however, this section did not reflect the change.

17.2.6.1 "Number second paragraph "6" and reformat bullets under this paragraph from "1 & 2" to "a & b". This is editorial only to match the new format and eliminate redundant numbering under same section.

LDCR-SA-2004-36, EVAL-2001-002847-05 (CBC):

EVAL-2001-002847-05 (LDCR-SA-2004-036) / Sections 1.2.2.3.7, 6.4.1.2, 6.4.2.2, 6.4.2.3, 6.4.2.4, T6.4-1 (Sheet 1), 9.4.1.1, 9.4.1.2, 9.4.1.3, 9.4.1.4, T9.4-10 (Sheet 1 - item 2.a) (Sheet 2 - items 2.c, 2.e) (Sheet 3 -item 2.f), 15.6.5.4: Per FSAR Section 9.4 and 6.5, the basic design functions of the inlet dampers associated with each Train of Control Room (CR) HVAC ventilation system are to: (1) open and let air into the CR via the ventilation system and to (2) close to isolate the CR ventilation system from outside air. The existing dampers do not perform this 2nd function as readily as desired as tracer gas testing has demonstrated some filtered in-leakage occurs when the ESF systems are lined up to operate in emergency modes.

Replacing the existing dampers with bubble tight dampers will minimize the amount of filtered in-leakage into the CRE when operating in emergency modes and one of the Trains of CR HVAC is shutdown. This activity does not change the original design functions of the individual dampers and thus, will not change the design function of the CR HVAC System.

LDCR-SA-2004-036 updates the FSAR providing information for the new type of dampers and their associated design functions of ensuring the assumptions made in the dose analyses remain valid and in accordance with the original design function of the CR HVAC system.

EVAL-2001-002847-05 (LDCR-SA-2004-036) / Section 9.4.1.1: The word "inoperative" is replace with the word "inoperable." This is an editorial clarification.

EVAL-2001-002847-05 (LDCR-SA-2004-036) / Section 9.4.1.2: Replace "resistand with "resistan." This is an editorial correction.

EVAL-2001-002847-05 (LDCR-SA-2004-036) / Section 9.4.1.2: Replace "The main toilet" with "The toilet". This is an editorial clarification.

FSAR Amendment 100, Supplement b

LDCR-SA-2004-36, EVAL-2001-002847-05 (CBC) (continued):

EVAL-2001-002847-05 (LDCR-SA-2004-036) / Section9.4.1.3: Per FSAR Section 9.4.

LDCR-SA-2005-18, EVAL-2001-002657-03 (CBC):

Section 9.1.4.2.3(14): Revise Section 9.1.4.2.3(14), "Fuel Handling System / System Description / Component Description / Polar Crane," to clarify operation of the polar crane during MODES 1 through 4. Previously the FSAR indicated the crane remained essentially in the parked position during MODES 1 and 2 and the crane may be operated through its full 360 degree travel range during MODES 3 and 4. The revised discussion indicates the crane remains essentially in the parked position during MODES 1 through 4 unless maintenance is being performed on it. The Containment Access Rotating Platform is allowed to be used for maintenance on (including modifications to) the Polar Crane in MODES 1 through 4. During Polar Crane maintenance the polar crane may be operating through its full 360 degree travel range. This change is acceptable since any maintenance activities performed on the polar crane (including its movement) during MODES 1 through 4 are subject to risk assessments performed in accordance with the maintenance rule 10CFR50.65(a)(4).

The phrase "...any safety related SSC (e.g., Hydrogen Recombiner)." is changed to "... any safety related SSC." This excessive level of detail is not required for the FSAR.

Change the phrase "No load will be carried during MODES 1-4." to "No load will be carried by the polar crane during MODES 1-4." This consistent with the changes above.

The phrase "A compensatory measure (operator, in direct communication..." is changed to "A compensatory measure (a dedicated person, ...". An operator is not required for this administrative function.

LDCR-SA-2005-20, EVAL-2005-003243-01 (CBC):

Section 9.1.4.2.3: EVAL-2005-003243-01 / LDCR-SA-2005-20: Changes description in FSAR Section 9.1.4.2.3, "Fuel Handling System / System Description / Component Description / Reactor Vessel Stud Tensioner," by revising the discussion of the stud tensioning process for the reactor vessel head. The method used to pretension the reactor vessel head studs has changed. The requirement that the tensioning method prevent unequal loading of the studs and excessive stress in the flange region remains unchanged, therefore the change is acceptable. The old description contained excessive detail and has been deleted and a statement added that an approved sequence is used to tension the studs.

LDCR-SA-2005-28, EVAL-2004-002831-04 (CBC):

This change to Section 9.1.4.2.3(4) allows the new fuel elevator to also be used in the reconstitution of irradiated fuel. The new fuel elevator will have mechanical stops added that will not adversely affect the elevator's design and its electrical limit switches readjusted to ensure the 10 feet of water shielding is maintained during fuel

LDCR-SA-2005-28, EVAL-2004-002831-04 (CBC) (continued):

reconstitution. During normal operation the mechanical stops are removed and electrical limits will be reset, allowing the new fuel elevator to lower new fuel into the cask pit area. The addition of the mechanical and electrical stops will have no adverse impact on any SSC design function.

LDCR-SA-2005-31, EVAL-2001-001914-14 (CBC):

Section 8.2.1: In Section 8.2.1, revise the stated use of Spare Startup Transformer XST1/2 to be a physical replacement of Startup Transformer XST2 instead of being energized at its storage location west of the Turbine Building from the 345 kV line by closing a normally open motor-operated air-switch. The description of the physical relocation of Spare Transformer XST1/2 and connection to the 138 kV line to XST1 has not changed. Spare Startup Transformer XST1/2 is presently located west of the Turbine Building across the access road under the 345 kV line feeding Station Service Transformer 1ST and Startup Transformer XST2. The Spare Transformer high voltage bushings are connected thorough a motor-operated air-switch to the 345 kV line. The air switch is locked open. The original intent was to use the Spare Transformer from this location in the event of a failure of Startup Transformer XST2 by building a 6.9 kV cable bus across the access road and into the TB for temporary connection into the plant 6.9 kV cable bus system. However, a more practical use of Spare Transformer XST1/2 in the event of a failure of XST2 is to physically move out XST2 and place the Spare Transformer on the XST2 pad.

This allows the 6.9 kV secondary cable bus to be directly connected to the Spare Transformer without modification of the secondary cable bus. The space between the 6.9 kV bushing phases and their height is exactly the same for both transformers. The only modification required is to identify the alternate terminations for power, control and protective relay circuits. FDA-2001-001914-16 is a paper change only contingency modification providing alternate connections for Spare Transformer XST1/2 when physically replacing Startup Transformer XST2. This change to the FSAR more accurately describes the use of Spare Transformer XST1/2 in the physical replacement of XST2.

LDCR-SA-2004-19, EVAL-1999-003133-08 (CBC):

Section 3.3.2.2, 9.4C.2, and 9.4C.3: Revise description of Section 3.3.2.2 , " Tornado Protection Design Features - Tornado Pressure Relief Dampers In Interior Fire Walls", Section 9.4C.2 " Main Steam and Feedwater Piping Area Ventilation System", Section 9.4C.3 " Electrical Area (Safeguards) Ventilation System"

The Safeguards Electrical Area Ventilation System (EAVS) may not be able to maintain the minimum design and licensing limiting temperature in the MS/FW Pipe & Electrical Area HVAC Room rooms 1-104 and 2-104, during extreme wintertime conditions. The Main Steam and Feedwater (MS & FW) Area Ventilation System may not be able to maintain the maximum design and licensing limiting temperature in the Main Steam and

LDCR-SA-2004-19, EVAL-1999-003133-08 (CBC) (continued):

Feedwater Pipe Penetration Areas, during extreme summertime conditions. [Reference SMF-1999-003133]

This modification, FDA-1999-003133-01-02, connects the EAVS and MS &FW Area Ventilation Systems in order to enhance the efficiency of each ventilation system in performing its design function. The modified ventilation systems establish new EAVS exhaust air flow paths through the areas served by EAVS. EAVS tempered exhaust air is drawn through a common stairwell, open doors and floor openings from the electrical areas to the fresh air intake of the MS & FW Area Ventilation System air handling unit located in the Mechanical Equipment Room on Elevation 852'-6".

This modification does not result in an adverse impact on the design function both stated and implicit for those plant components, systems, programs and licensing basis requirements impacted by this activity. The activity does not result in an adverse impact to the design functions of SSC's as described in the FSAR.

LDCR-SA-2005-23, EVAL-2002-000747-01 (CBC):

Table 3.9B-10: EVAL-2002-000747-01 (LDCR-SA-2005-23) Revise Table 3.9B-10, "Active Valves" to clarify the method of actuation for LV-4500, LV-4501, and LV-4500-1. Section 9.2.2.5.3 of the FSAR notes the manual operation of the valves as well as the safety function. This is a trivial administrative clarification of a supporting section.

LDCR-SA-2004-43, EVAL-2004-000038-01 (CBC):

Figures 3.6B-64-1 and 3.6B-64-2: EVAL-2004-000038-01 (LDCR-SA-2004-043): Inactive pipe whip restraint RC-1-007-902-C47W is being deleted and removed from the plant. This LDCR will update the Restraint Location drawing in the FSAR to reflect the facilities configuration. The removal of a non-functioning (i.e., abandoned) pipe whip restraint does not result in an adverse impact on the ability of the plant SSC's to mitigate the consequences of a postulated accident. This restraint presents obstructions to outage maintenance activities that result in increased dose and the potential for injury.

LDCR-SA-2005-15, EVAL-2002-004206-03 (RJK):

In FSAR Section 3.6B.2.5.2 the design pressure for the portion of the Unit 1 Safety Injection (SI) System from and including the SI Pumps up to the outside containment isolation valves is being increased from 1750 psig to 1860 psig to resolve discrepancies identified during an NRC SSD&PC Inspection in 2002. The discrepancies identified were that the SI pump shutoff head identified in the FSAR and DBD was less than the maximum shutoff head shown on the vendor certified pump curves.

The Unit 1 system design pressure has been established at 1860 psig to bound the maximum SI pump discharge pressure that could exist following a LOCA and while in the recirculation mode. The system piping and components have been re-evaluated for this higher design pressure. In Table 6.3-2 the relief valve set points have been changed to

FSAR Amendment 100, Supplement b

LDCR-SA-2005-15, EVAL-2002-004206-03 (RJK) (continued):

1860 psig, i.e., just above the maximum postulated SI pump discharge pressure, to prevent their opening while in the recirculation mode. Increasing the relief valve set points from 1750 psig to 1860 reduces the probability that the valves will open while the SI system is operating in the recirculation mode and, therefore, provides greater assurance that they can perform their function as a boundary for a closed system outside containment.

In Table 6.3-1 the Unit 1 system design pressure has been established at 1860 psig to bound the maximum SI pump discharge pressure that could exist following a LOCA and while in the recirculation mode. The system piping and components have been re-evaluated for this higher design pressure.

LDCR-SA-2005-2, EVAL-2001-001255-12 (TJE):

Sections 8.3.1.1.11.1 and 8.3.1.1.11.2: The changes in sections 8.3.1.1.11.1 and 8.3.1.1.11.2 provide a description of the new Unit 1 Train B upgraded digital excitation/voltage regulator that will be consistent with the previous digital excitation/voltage regulator modifications on Unit 1 Train A and Unit 2 Trains A and B. See Markup in EVAL-2001-001255-12.

NOTE: The DG Exciter/Voltage Regulator Upgrade/replacement will replace the obsolete Exciter/Voltage Regulator for the Emergency Diesel Generator with a new digital-controlled exciter/voltage regulator which will be able to be serviced and replaced for the next 20 years.

Section 8.3.1.2.1, Part J: The changes in section 8.3.1.2.1 Part J, dealing with Compliance with the NRC Regulatory Guide 1.75, provides a description of the new Unit 1 Train B upgraded digital multifunction protective device that will be consistent with the previous digital multifunction protective device modifications on Unit 1 Train A and Unit 2 Trains A and B. See Markup in EVAL-2001-001255-12.

NOTE: The DG Exciter/Voltage Regulator Upgrade/replacement will replace the obsolete Exciter/Voltage Regulator for the Emergency Diesel Generator with a new digital-controlled exciter/voltage regulator which will be able to be serviced and replaced for the next 20 years.

LDCR-SA-2005-7, EVAL-2001-002847-10 (TJE):

Section 8.3.1.1.11.2: Currently, in section 8.3.1.1.11.2 under Protection Systems the sentence reads, "The valve motors are protected against sustained fault conditions by circuit breakers if full load currents are equal to or greater than 0.66 amps, and by fused disconnect switches if full load currents are less than 0.66 amps."

Change the sentence to read, "The valve motors are protected against sustained fault conditions by circuit breakers. The valve motors may be protected by fused disconnect switches if full load currents are less than 1.3 amps."

LDCR-SA-2005-7, EVAL-2001-002847-10 (TJE) (continued):

This change does not affect the ability of the MOVs to perform their required safety functions. The alternate design allowance for fuse to be used for MOV circuit protection is limited to only a small set of MOVs, originally provided with fuse protection by the MCC vendor, to further reduce the exposure of fuse blowing. As such the alternate design allowance for fuse to be used for MOV circuit protection is acceptable.

Section 8.3.1.2.1.6, Part c.1: Currently, in section 8.3.1.2.1.6 part c.1 the sentence reads, "For all starter circuits feeding motors having full load currents equal to or greater than 0.66A, and for all feeder circuits, a circuit breaker and starter with a thermal overload relay as primary protection in series with a circuit breaker as backup protection or two circuit breakers in series for primary and backup protection are provided in each circuit."

The sentence will be modified to read, "For all starter circuits feeding motors and for all feeder circuits, primary protection is generally provided with a circuit breaker and/or starter with a thermal overload relay in series with a circuit breaker as backup protection or two circuit breakers in series for primary and backup protection are provided in each circuit."

Section 8.3.1.2.1.6, part c.2: Additionally, section 8.3.1.2.1.6 part c.2 reads, "For all starter circuits feeding motors having full load currents less than 0.66A, primary and backup protection is provided by means of a fused disconnect switch and a circuit breaker respectively."

The sentence in c.2 will be modified to read, "For starter circuits feeding motors having full load currents less than 1.3 A, primary and backup protection may be provided by means of a fused disconnect switch and a circuit breaker respectively."

This change does not affect the ability of the MOVs to perform their required safety functions. The alternate design allowance for fuse to be used for MOV circuit protection is limited to only a small set of MOVs, originally provided with fuse protection by the MCC vendor, to further reduce the exposure of fuse blowing. As such the alternate design allowance for fuse to be used for MOV circuit protection is acceptable.

LDCR-SA-2005-10, EVAL-2000-000526-5 (TJE):

Figure 8.3-16: In FSAR Figure 8.3-16, removed unit differences in lower left hand table under titles "NORMAL" and "TOTAL".

Note: The fiber optic modules are installed to support the Wireless LAN Extension into the Unit 1 Containment Building and to support the new Radiation Protection Camera LAN system. The fiber optic modules will not impact the other circuits in the penetrations. FSAR Sections 8.3.1.4.4, 8.3.1.4.5 and 8.3.1.4.10 all state there is no separation required between Non-1E fiber circuits and 1E circuits. Fiber Optic circuits will not interfere with 125 Vdc or 120 Vac circuits in Type 3 Penetrations, nor will the control circuits interfere with the fiber optic circuits. Therefore, there is no concern that the fiber optic will cause a

LDCR-SA-2005-10, EVAL-2000-000526-5 (TJE) (continued):

fire that could damage the other circuits nor will the fiber impair the operation of any other circuits.

LDCR-SA-2005-19, EVAL-2003-001365-14 (TJE):

Section 8.2.1: In section 8.2.1, 1.) add the following sentence to the end of the 11th paragraph, "The network transmission line terminals of 345-kV switchyard are provided with a third set of relay protection to enhance the reliability of transmission system fault isolation."

In section 8.2.1, 2.) add the following sentence to the end of the 12th paragraph, "A separate 125-VDC system is provided for the third set of relay protection in the 345-kV switchyard."

Note: The third scheme of relay protection can provide this last resort backup protection locally, thereby minimizing the impact on the system. This will increase the availability and reliability of the 345kV offsite source for CPSES safety related buses during 345kV system faults.

LDCR-SA-2005-25, EVAL-2002-003739-1 (TJE):

Section 1A(B): In section 1A(B), under the Discussion of RG 1.75, the proposed changes will change the wording "...upstream circuit breakers..." to "...upstream circuit protective device..." in the first sentence of the second paragraph. Currently, the sentence reads as follows:

"Regulatory Position C.1 Non Class 1E power or control circuits may be isolated from their Class 1E power source by two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit breaker, and the circuit breakers periodically tested. Non safety instrument circuits powered from distribution panels 1PC1, 1PC2, 1PC3, and 1PC4 will have a non safety circuit breaker or fuse connected in series with the panel circuit breaker."

The sentence will be revised to read:

"Regulatory Position C.1 Non Class 1E power or control circuits may be isolated from their Class 1E power source by two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit protective device, and the circuit breakers periodically tested. Non safety instrument circuits powered from distribution panels 1PC1, 1PC2, 1PC3, and 1PC4 will have a non safety circuit breaker or fuse connected in series with the panel circuit breaker."

Note: The FSAR descriptions of, breakers and starters tripped by SIAS as an isolation device, and a single fuse as an isolation device for NSSS protection system, are being revised to include the coordination requirement with the upstream protective device. This is a clarification change to improve the readers under standing of an isolation device.

LDCR-SA-2005-25, EVAL-2002-003739-1 (TJE) (continued):

Section 8.3.1.2.1.7, part a: Currently, section 8.3.1.2.1.7 under “a. Power Circuits” reads,

"Power Circuits

The following types of devices are used in the CPSES design for isolation of power circuits:

1) Circuit breaker tripped by a safety injection signal.

2) Starter contactor opened by a safety injection signal.

3) Two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit breaker, and the circuit breakers periodically tested."

The proposed changes will revise this section to read,

"Power Circuits

The following types of devices are used in the CPSES design for isolation of power circuits:

1) Circuit breaker coordinated with the upstream protective device and tripped by a safety injection signal.

2) Circuit protective device of a starter coordinated with the upstream protective device and starter contactor opened by a safety injection signal.

3) Two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit protective device, and the circuit breakers periodically tested."

Note: The FSAR descriptions of, breakers and starters tripped by SIAS as an isolation device, and a single fuse as an isolation device for NSSS protection system, are being revised to include the coordination requirement with the upstream protective device. This is a clarification change to improve the readers under standing of an isolation device.

Section 8.3.1.2.1.7, part b: Currently, section 8.3.1.2.1.7 and 8.3.1.2.1.7 under “b. Control and Instrumentation Circuits” parts 5 and 6 read,

"5) Two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit breaker, and the circuit breakers periodically tested, provide isolation between Class 1E and non Class 1E circuits.

6) Within the NSSS protection system cabinets, a fuse or breaker is an isolation device to prevent malfunctions in Non-Class 1E portions of a circuit from causing unacceptable influences on the Class 1E function of the circuit (Reference 48). This exception to the

LDCR-SA-2005-25, EVAL-2002-003739-1 (TJE) (continued):

isolation devices described in the preceding paragraph (5), is applicable only to the NSSS circuit design by Westinghouse. See Section 7.1.2.2.1."

The proposed changes will revise these two paragraphs to read,

"5) Two circuit breakers, two fuses or a breaker and a fuse in series, both coordinated with an upstream circuit protective devise, and the circuit breakers periodically tested, provide isolation between Class 1E and non Class 1E circuits.

6) Within the NSSS protection system cabinets, a fuse or breaker, coordinated with the upstream protective devise, is an isolation device to prevent malfunctions in Non-Class 1E portions of a circuit from causing unacceptable influences on the Class 1E function of the circuit (Reference 48). This exception to the isolation devices described in the preceding paragraph (5), is applicable only to the NSSS circuit design by Westinghouse. See Section 7.1.2.2.1."

Note: The FSAR descriptions of, breakers and starters tripped by SIAS as an isolation device, and a single fuse as an isolation device for NSSS protection system, are being revised to include the coordination requirement with the upstream protective device. This is a clarification change to improve the readers under standing of an isolation device.

LDCR-SA-2005-32, EVAL-1999-002406-2 (TJE):

Section 1A(B): 1.) In 1A(B) under the Discussion of RG 1.75 after 36th paragraph, insert the following words:

"Physical separation between non-Class 1E and Class 1E circuits or non-Class 1E and associated Class 1E circuits inside the Diesel Generator Engine Control Panels CP1/2-MEDGEE-01A/02A is not required based on the analysis provided in Section 8.3.1.4.5."

Section 8.3.1.4.5: 2.) Insert the following words after the last paragraph of sections 8.3.1.4.5:

"Separation between non-Class 1E and Class 1E circuits or non-Class 1E and associated Class 1E circuits in the Diesel Generator Engine Control Panels CP1/2-MEDGEE-01A and 02A, listed in Table 8.3-10, is not required based on the following analysis. These panels are re-classified as Multi-Train cabinets. Both the AC and DC circuits for the Diesel Generator Engine Control Panels are supplied by a Class 1E power supply. The non-Class 1E circuits in the control panels are electrically isolated by an approved design having two circuit breakers in series for electrical isolation to assure that the Class 1E power source is not adversely affected by a fault in the non-Class 1E circuit.

Internal Train Separation Exemption:

Per Regulatory Guide 1.75, non-Class 1E circuits shall be physically separated from Class 1E circuits and non-Class 1E circuits shall be physically separated from associated

LDCR-SA-2005-32, EVAL-1999-002406-2 (TJE) (continued):

Class 1E circuits by the minimum separation requirements. All cables installed in the Diesel Generator Engine Control Panels are flame retardant and meet IEEE-383 flame test requirement. Any non-Class 1E cable fault, short circuit, short to ground, or over load condition will be isolated by an isolation breaker and will not cause the cable temperature to exceed normal operating temperature. The non-Class 1E cables inside the control panel will not be degraded or will not reach the temperature that can affect the Class 1E or associated Class 1E circuits. Open non-Class 1E circuit, have no impact on Class 1E or associated Class 1E circuits. The non-Class 1E cables when maintained within their rated temperature, have no adverse impact on adjacent Class 1E or associated Class 1E cables with no physical separation because the Class 1E and associated Class 1E cables also have the same temperature ratings.

Therefore, the adequacy of the safety-related circuits to perform their safety function is maintained with no separation between non-Class 1E and Class 1E circuits or non-Class 1E and associated Class 1E circuits inside the Diesel Generator Engine Control Panels. Any faults in the non-Class 1E circuits will not compromise the safety function of the Class 1E or associated Class 1E circuits internal to the Diesel Generator Engine Control Panels. Therefore, there is no need for internal electrical separation requirements within Diesel Generator Engine Control Panels CP1/2-MEDGEE-01A and 02A."

Table 8.3-10, sheet 9: 3.) Add the next four lines at the end of TABLE 8.3-10 and before the Notes:

"CP1-MEDGEE-01A DIESEL GENERATOR 1-01 DIESEL ENGINE CONTROL PANEL -- 18

CP1-MEDGEE-02A DIESEL GENERATOR 1-02 DIESEL ENGINE CONTROL PANEL -- 18

CP2-MEDGEE-01A DIESEL GENERATOR 2-01 DIESEL ENGINE CONTROL PANEL -- 18

CP2-MEDGEE-02A DIESEL GENERATOR 2-02 DIESEL ENGINE CONTROL PANEL -- 18"

Table 8.3-10, sheet 10: 4.) Add Note "18" to end of Table 8.3-10.

"18. Equipment supplied by Delaval Engine & Compressor has been analyzed and is exempt from internal separation requirement. (Reference FSAR Section 8.3.1.4.5.)"

LDCR-SA-2006-6, EVAL-2006-000595-1 (TJE):

Section 1.6: 1.) Currently in section 1.6, the first sentence of the first paragraph reads, "Table 1.6-1 lists topical reports and other licensing documents which provide information additional to that provided in this FSAR and have been filed separately with the Nuclear Regulatory Commission (NRC) in support of the CPSES operating licenses."

LDCR-SA-2006-6, EVAL-2006-000595-1 (TJE) (continued):

Delete the words, "topical reports and other licensing" from this sentence.

The sentence will then read, "Table 1.6-1 lists documents which provide information additional to that provided in this FSAR and have been filed separately with the Nuclear Regulatory Commission (NRC) in support of the CPSES operating licenses."

2.) Currently in section 1.6, the last sentence of the first paragraph reads, "Later revisions to these documents are not part of the FSAR unless they are incorporated into this table."

Add the following words to the end of this sentence, "(except for the Technical Requirements Manual (including TRM Bases) and the Technical Specifications Bases)"

The sentence will then read, "Later revisions to these documents are not part of the FSAR unless they are incorporated into this table (except for the Technical Requirements Manual (including TRM Bases) and the Technical Specifications Bases)"

Note:

This change is consistent with Technical Specification 5.5.14 and 5.5.17 which allow changes to the TRM and TSB under 10CFR50.59. Appendix A.4.3 of NEI-98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, allows these type of documents to be incorporated into the FSAR by reference. NEI-98-03, Revision 1, is endorsed by NRC Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)." This is also consistent with FSAR Section 16.2. This will help eliminate confusion and allow certain changes (e.g., reformatting, consistency, and others as defined in NEI-03, Revision 1) to be excluded from 10CFR50.59 review requirements.

Table 1.6-1, sheet 2: Replace the words in the title of Table 1.6-1 from "TOPICAL REPORTS" to "DOCUMENTS" on Sheet 2.

Note:

This change is consistent with Technical Specification 5.5.14 and 5.5.17 which allow changes to the TRM and TSB under 10CFR50.59. Appendix A.4.3 of NEI-98-03, “Guidelines for Updating Final Safety Analysis Reports,” Revision 1, allows these type of documents to be incorporated into the FSAR by reference. NEI-98-03, Revision 1, is endorsed by NRC Regulatory Guide 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e).” This is also consistent with FSAR Section 16.2. This will help eliminate confusion and allow certain changes (e.g., reformatting, consistency, and others as defined in NEI-03, Revision 1) to be excluded from 10CFR50.59 review requirements.

Table 1.6-1, sheet 3: 1.) Replace the words in the title of Table 1.6-1 from "TOPICAL REPORTS" to "DOCUMENTS" on Sheet 3.

LDCR-SA-2006-6, EVAL-2006-000595-1 (TJE) (continued):

2) Add the following two Reports and Reference Sections to the end of Table 1.6-1 on Sheet 3:

Technical Requirements Manual (TRM) for Comanche Peak Steam Electric Station Units 1 and 2	All Sections
Technical Specifications Bases for Comanche Peak Steam Electric Station Units 1 and 2	All Sections

Note:

This change is consistent with Technical Specification 5.5.14 and 5.5.17 which allow changes to the TRM and TSB under 10CFR50.59. Appendix A.4.3 of NEI-98-03, “Guidelines for Updating Final Safety Analysis Reports,” Revision 1, allows these type of documents to be incorporated into the FSAR by reference. NEI-98-03, Revision 1, is endorsed by NRC Regulatory Guide 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e).” This is also consistent with FSAR Section 16.2. This will help eliminate confusion and allow certain changes (e.g., reformatting, consistency, and others as defined in NEI-03, Revision 1) to be excluded from 10CFR50.59 review requirements.

LDCR-SA-2006-7, EVAL-2006-000627-2 (DWS):

Section 13.3B-1: Make organizational change "Senior Vice President and Principal Nuclear Officer" to "Senior Vice President and Chief Nuclear Officer"

TMI - I.C-3: Make organizational change. Change "Vice President, Nuclear Operations" to "Site Vice President"

In 13.1.1.2.1, change "Vice President, Nuclear Operations" to "Site Vice President" wherever it appears. Change "Vice President, Nuclear Engineering" to "Vice President, Nuclear Engineering and Support" wherever it appears. Realign reporting requirements for some managers. Delete obsolete manager titles and include new titles as appropriate.

Section 13.1A: Change "Vice President, Nuclear Operations" to "Site Vice President" wherever it appears. Change "Vice President, Nuclear Engineering" to "Vice President, Nuclear Engineering and Support" wherever it appears. Delete obsolete manager titles and include new titles as appropriate. Add QA manager title and resume.

In Figure 13.1-2, change "Vice President, Nuclear Operations" to "Site Vice President" . Change "Vice President, Nuclear Engineering" to "Vice President, Nuclear Engineering and Support" wherever it appears. Realign reporting requirements for some managers. Reinstate "Plant Manager" position.

LDCR-SA-2006-7, EVAL-2006-000627-2 (DWS) (continued):

In Figure 13.1-3, change "Vice President, Nuclear Operations" to "Site Vice President" and reinstate "Plant Manager" position.

In Table 13.1-1, change "Vice President, Nuclear Operations" to "Site Vice President" and add QA Manager

LDCR-SA-2006-20, EVAL-2006-000798-2 (DWS):

Section 13.3B - Revise ORC responsibility to assess the effectiveness of the fire protection program through review of periodic audits as discussed in FSAR 17.2.18 and corrected management title.

Audits have been moved to Section 17.2.18 and are conducted by the Nuclear Overview organization.

Section 13.4 - Revised to correct wording in FSAR from NRC audits to NRC inspection reports and to reflect changes made in FSAR Section 17.2.

LDCR-SA-2004-47, EVAL-2004-001171-2 (MJR):

FSAR Table 6.2.4-1, "Containment Isolation Valving Application," footnote a) for Table item 129 is revised to clarify that valve SF-0001 associated with the Fuel Transfer Tube penetration is a non-safety related valve that has no direct containment isolation function during MODES 1 through 4. This valve is normally closed in MODES 1-4. The valve may be opened in MODES 1-4 except when the Fuel Transfer Canal is flooded to maintain the Fuel Transfer Flange as leak rate tested. Opening the valve allows for equipment testing and to support preparations for refueling operations. (This change is consistent with clarifications added to DBD-ME-013 and DBD-ME-080 per approved and implemented FDA-2004-001171-01-00.)

LDCR-SA-2003-47 (MJR):

Appendix 1A(B), Regulatory Guide 1.75, is revised to add the Unit 1 and 2 Emergency Diesel Generators to the discussion of the Partial Discharge Monitor Bus Couplers. PDM Bus Couplers were installed on EDGs per FDA-2001-001255-02. The previous reference to Unit 2 only for Component Cooling Water motor feeders is deleted to reflect that PDM Bus Couplers have now been installed on both Unit 1 and 2 Component Cooling Water motors per FDA-2000-000127.

Appendix 1A(B), Regulatory Guide 1.131, is revised to add the Unit 1 and 2 Emergency Diesel Generators to the discussion of the Partial Discharge Monitor Bus Coupler jumpers. PDM Bus Couplers were installed on EDGs per FDA-2001-001255-02.

Section 8.3.1.2.1.7, Onsite Power Systems, Compliance, RG 1.75 and IEEE-354, is revised to add the Unit 1 and 2 Emergency Diesel Generators to the discussion of the Partial Discharge Monitor Bus Coupler jumpers. PDM Bus Couplers were installed on

LDCR-SA-2003-47 (MJR) (continued):

EDGs per FDA-2001-001255-02, and specific component tag numbers and wire size details are deleted as below the level of detail required for the FSAR description. Also, the Component Cooling Water Pump Motors were added to the discussion for completeness. PDM Bus Couplers were installed on CCWP motors per FDA-2000-000127.

Section 9.2.2.2 System Description is revised to delete the previous reference to Unit 2 for the CCW pump motors to reflect that PDM Bus Couplers have now been installed on both Unit 1 and 2 Component Cooling Water pump motors per FDA-2000-000127.

Section 9.5.1.6.2, Fire Protection System, Justification for Items of Noncompliance to Appendix A to Branch Technical Position APCSB 9.5-1, Guideline D.3.f, is revised to add the Unit 1 and 2 Emergency Diesel Generators to the discussion of the Partial Discharge Monitor Bus Coupler jumpers. PDM Bus Couplers were installed on EDGs per FDA-2001-001255-02. Also, specific component tag numbers are deleted as below the level of detail required for the FSAR description.

Table 8.3-3, "Failure Mode and Effect Analysis for Auxiliary AC Power System," is revised to add items 15C, 15D, 16C, and 16D for the Partial Discharge (PD) bus couplers installed on the Emergency Diesel Generators per FDA-2001-001255-05, and to add an abbreviation note to define "partial discharge (PD)."

LDCR-SA-2004-22, EVAL-2004-002167-1 (GLM):

Revise Section 9.2.8 to be consistent with the current approved waste effluent release permit. This includes changes to reflect current terminology as well as changes to bring the FSAR Section in line with the current wastewater discharge permit.

Section 9.2.8 of the FSAR documents the Design Basis and System Description of the Waste Management System (WMS) at CPSES. The design function of the Waste Management System is the collection, retention, treatment and discharge of normally non-radioactive wastewaters. The system does this in accordance with the Texas Commission on Environmental Quality (TCEQ) and the Texas Pollutant Discharge Elimination System (TPDES) permits. The WMS collects wastewaters for processing in either the Low Volume Waste (LVW) or Co-Current Waste (COW) treatment facilities. The goal of most of the changes is to provide the description of the major components in the system and to reiterate that the release of effluents is done within the limitation of our Discharge Permit. Specifically, the wording changes involved in this submittal are administrative in nature. The regulatory agency functions of the agencies in regard to CPSES have remained the same, however, the names of the agencies have changed. The Texas Natural Resource Conservation Commission (TNRCC) is now the Texas Commission on Environmental Quality (TCEQ). The National Pollutant Discharge Elimination System (NPDES) is now the Texas Pollutant Discharge Elimination System (TPDES). While this change request was being processed, it was determined that the description of the systems should be updated to be more consistent with the wording of the current approved discharge permit and some change was necessary to outline some

LDCR-SA-2004-22, EVAL-2004-002167-1 (GLM) (continued):

current operating practices. This submittal also includes corrections in system terminology in the descriptions. The LVW facilities required more change than did the COW facility descriptions. For the most part, the facility descriptions remain unchanged although the wording has been reformatted to align better with the Discharge Permit. The changes requested better describe the use of the facilities and reinforce some detail regarding both use and operation of the facilities which fall within applicable TCEQ permit limitations.

Revise Section 9.2.8 to be consistent with the current approved waste effluent release permit. This includes changes to reflect current terminology as well as changes to bring the FSAR Section in line with the current wastewater discharge permit.

Section 9.2.8 of the FSAR documents the Design Basis and System Description of the Waste Management System (WMS) at CPSES. The design function of the Waste Management System is the collection, retention, treatment and discharge of normally non-radioactive wastewaters. The system does this in accordance with the Texas Commission on Environmental Quality (TCEQ) and the Texas Pollutant Discharge Elimination System (TPDES) permits. The WMS collects wastewaters for processing in either the Low Volume Waste (LVW) or Co-Current Waste (COW) treatment facilities. The goal of most of the changes is to provide the description of the major components in the system and to reiterate that the release of effluents is done within the limitation of our Discharge Permit. Specifically, the wording changes involved in this submittal are administrative in nature. The regulatory agency functions of the agencies in regard to CPSES have remained the same, however, the names of the agencies have changed. The Texas Natural Resource Conservation Commission (TNRCC) is now the Texas Commission on Environmental Quality (TCEQ). The National Pollutant Discharge Elimination System (NPDES) is now the Texas Pollutant Discharge Elimination System (TPDES). While this change request was being processed, it was determined that the description of the systems should be updated to be more consistent with the wording of the current approved discharge permit and some change was necessary to outline some current operating practices. This submittal also includes corrections in system terminology in the descriptions. The LVW facilities required more change than did the COW facility descriptions. For the most part, the facility descriptions remain unchanged although the wording has been reformatted to align better with the Discharge Permit. The changes requested better describe the use of the facilities and reinforce some detail regarding both use and operation of the facilities which fall within applicable TCEQ permit limitations. The second thru fifth paragraphs were deleted since they provide both duplicate information and more detail than is necessary to describe the design volumes in the settling ponds. The sixth paragraph was deleted since it is redundant to information in a new paragraph. This paragraph details periodic testing of Condensate Polishing wastes. This information is accurate but is already covered by the statements in the second paragraph which indicates that releases are controlled via the ODCM program. Typically wastes sent to the Low Volume Waste Management System do not include radioactive wastes. Although there are accident scenarios which postulate such releases, there is no normal release of such wastes to the Low Volume Waste System. Releases

LDCR-SA-2004-22, EVAL-2004-002167-1 (GLM) (continued):

are controlled via the ODCM limits as well as the limits established by the Release Permit.

Revise Section 9.2.8 to be consistent with the current approved waste effluent release permit. This includes changes to reflect current terminology as well as changes to bring the FSAR Section in line with the current wastewater discharge permit.

Section 9.2.8 of the FSAR documents the Design Basis and System Description of the Waste Management System (WMS) at CPSES. The design function of the Waste Management System is the collection, retention, treatment and discharge of normally non-radioactive wastewaters. The system does this in accordance with the Texas Commission on Environmental Quality (TCEQ) and the Texas Pollutant Discharge Elimination System (TPDES) permits. The WMS collects wastewaters for processing in either the Low Volume Waste (LVW) or Co-Current Waste (COW) treatment facilities. The goal of most of the changes is to provide the description of the major components in the system and to reiterate that the release of effluents is done within the limitation of our Discharge Permit. Specifically, the wording changes involved in this submittal are administrative in nature. The regulatory agency functions of the agencies in regard to CPSES have remained the same, however, the names of the agencies have changed. The Texas Natural Resource Conservation Commission (TNRCC) is now the Texas Commission on Environmental Quality (TCEQ). The National Pollutant Discharge Elimination System (NPDES) is now the Texas Pollutant Discharge Elimination System (TPDES). While this change request was being processed, it was determined that the description of the systems should be updated to be more consistent with the wording of the current approved discharge permit and some change was necessary to outline some current operating practices. This submittal also includes corrections in system terminology in the descriptions. The LVW facilities required more change than did the COW facility descriptions. For the most part, the facility descriptions remain unchanged although the wording has been reformatted to align better with the Discharge Permit. The changes requested better describe the use of the facilities and reinforce some detail regarding both use and operation of the facilities which fall within applicable TCEQ permit limitations. The second paragraph regarding Chemical Sump Wastes was also deleted. This paragraph indicates that Chemical Sump Wastes are being discharged to the Equalization Basin and in turn to the Natural Neutralization Basin, Neutralization Clearwell with an ultimate discharge and/or commingling with LVW Pond effluents. These elements and processing within the LVW facilities are no longer in normal practice. Currently, effluents from the Chemical Sumps are processed together with the other plant wastewaters via the Separation Surge Basin, API Oil Separator and Settling Basins. Descriptions regarding the use of the Equalization Basin lineup is no longer relevant or necessary based on the current normal operating lineups or practices.

Revise Section 9.2.8 to be consistent with the current approved waste effluent release permit. This includes changes to reflect current terminology as well as changes to bring the FSAR Section in line with the current wastewater discharge permit.

LDCR-SA-2004-22, EVAL-2004-002167-1 (GLM) (continued):

Section 9.2.8 of the FSAR documents the Design Basis and System Description of the Waste Management System (WMS) at CPSES. The design function of the Waste Management System is the collection, retention, treatment and discharge of normally non-radioactive wastewaters. The system does this in accordance with the Texas Commission on Environmental Quality (TCEQ) and the Texas Pollutant Discharge Elimination System (TPDES) permits. The WMS collects wastewaters for processing in either the Low Volume Waste (LVW) or Co-Current Waste (COW) treatment facilities. The goal of most of the changes is to provide the description of the major components in the system and to reiterate that the release of effluents is done within the limitation of our Discharge Permit. Specifically, the wording changes involved in this submittal are administrative in nature. The regulatory agency functions of the agencies in regard to CPSES have remained the same, however, the names of the agencies have changed. The Texas Natural Resource Conservation Commission (TNRCC) is now the Texas Commission on Environmental Quality (TCEQ). The National Pollutant Discharge Elimination System (NPDES) is now the Texas Pollutant Discharge Elimination System (TPDES). While this change request was being processed, it was determined that the description of the systems should be updated to be more consistent with the wording of the current approved discharge permit and some change was necessary to outline some current operating practices. This submittal also includes corrections in system terminology in the descriptions. The LVW facilities required more change than did the COW facility descriptions. For the most part, the facility descriptions remain unchanged although the wording has been reformatted to align better with the Discharge Permit. The changes requested better describe the use of the facilities and reinforce some detail regarding both use and operation of the facilities which fall within applicable TCEQ permit limitations.

LDCR-SA-2004-44, EVAL-2001-000158-7 (GLM):

Revise Section 9.3.1 to reflect the following: 1) the word "Spare" is no longer applicable as the common compressors may be operated as the lead compressor for either unit. Therefore, spare is changed to "common", 2) each common and Unit 2 instrument air compressor has a dedicated trim cooler cooled by non-safety chilled water, 4) the Unit 1 CCW system remains in a series flow path for the unit 1 compressors. Therefore, a single trim cooler is used, 3) due to the elevated temperatures in the Turbine Building during the Summer months, a non-safety chilled water ventilation cooler is provided to duct cooled air into the compressor air inlet, ensuring intake parameters are within operational limits.

Revise Section 9.3.1 to reflect the following: Each instrument air dryer has two 100% capacity prefilters.

Revise Table 9.3-1 to reflect the following: On sheets 1 and 2, Part "A" is deleted as all 435 SCFM, rotary, two stage, oil free, water cooled air compressors have been removed from the plant.

LDCR-SA-2004-44, EVAL-2001-000158-7 (GLM) (continued):

Revise Table 9.3-1 to reflect the following: On sheet 3, revise the last 2 entries in the table to reflect that the exit cooling water temperature for the Common and Unit 2 compressors is 122 degrees F and the required cooling water flow for the Common and Unit 2 compressors is 40 gpm.

Revise Section 8.3.1.2.1 to reflect the following: The new instrument air compressor control panel CP2-CICACO-01B is Train C and some associated cables are terminated at the control panel without isolation devices.

Revise Table 8.3-10 to reflect the following: Add new instrument air compressor control panel CP2-CICACO-01B to the table.

Revise Table 8.3-11 to reflect the following: The Unit 1 power source and ID numbers are editorially corrected for instrument air compressors 1-01 and 1-02 and instrument air dryer control panel 1-02.

Revise Table 8.3-11 to reflect the following: Add CP2-CICACO-01, CP2-CICACO-02, CP2-CICYIA-01, and CP2-CICYIA-02 to the table.

LDCR-SA-2005-27, EVAL-2004-002128-4 (GLM):

Revise Section 9.2.3.1 to add description of nitrogen sparger that was installed in the Unit 1 RMWST per FDA-2004-002128-01. Adding nitrogen to the RMWST will improve the control of the dissolved oxygen concentration.

LDCR-SA-2006-8, EVAL-1999-000248-8 (GLM):

Revise Table 9.4-2 to add a note indicating that the temperatures in the DG area are maintained with the help of the DG keepwarm system.

The current design basis was established by Calculation 1-EB-302A-1 Rev. 5 "As built HVAC Calc Diesel Generator Area--Space Heat Gains, Space Heat Losses and Maximum and Minimum Temperatures - Unit 1. (dated 9/20/1989). The same design was confirmed in Unit 2 Calc. 2-HV-0009 Rev. 1. 1-EB-302-1 Rev. 5 on page 3 indicates the reason for Rev. 5 "Calculation Rev. 5 is a complete revision which takes credit for the D.G. "Keepwarm" system for space heating as per TU-Electric Letter No. OSE-89026 dated 9-4-89." The justification for this licensing basis change and the Engineering Basis for FDA-1999-000248-03 is Calculation 1-EB-302-1 Rev. 5. issued on 9/20/1989. The calculation demonstrates that the DG Jacket water 52.7 KW heater with the 27.5 KW Lube oil heater and the two installed 5 KW room heaters was more than enough heat to keep all portions of the Unit 1 DG rooms well above 40ºF. 1-EB-302-1 Rev 5 determined that the available safety margin is 56.8%. However, if we have equipment failure, or if the outside ambient temperature falls below the design basis 20ºF temperature or if we have excessive infiltration, supplemental heating and/or other compensatory measures may be required too maintain DG room temperature at or above 40ºF.

FSAR Amendment 100, Supplement b

LDCR-SA-2006-19, EVAL-2004-003769-19 (GLM):

Revise Section 9.3.1.2 to delete "to regulate flow" since, although it is a function of the valves, it is not a design function of the accumulators. Discussion of the flow regulation is moved to Section 10.4.9.2 which is referenced for clarification.

Clarification suggested by SMF-2004-003769 generated in response to NRC questions. There are no changes to the design function or the method of performing the design function. See FDA-2004-003769-01 for design impact assessments and additional details.

Revise Section 10.4.9.2 to reflect the following: Add clarification to incorporate information from Section 6.2.1, SSER 17, and CPSES-200402640. This places all pertinent information in one location.

Clarification suggested by SMF-2004-003769 generated in response to NRC questions. There are no changes to the design function or the method of performing the design function. See FDA-2004-003769-01 for design impact assessments and additional details.

LDCR-SA-2006-17, EVAL-2006-000809-1 (JCH):

Table 10.4-11 entry for chlorides is incorrect. Chloride value should be "10.0" vs "1.0" mg/l (max)

This appears to be a typo. DBD-ME-212 was changed from 1.0 to 10.0 mg/l by FDA-2001-002476-01 to match existing Chemistry procedures (which had been revised to match EPRI Closed cooling Water Chemistry Guideline 1007820). The FDA failed to identify the FSAR as impacted.

LDCR-SA-2006-21, EVAL-2006-002113-1 (DWS):

Section 13.1A: Added new plant manager's resume.

FSAR Amendment 101

LDCR-SA-2005-13, EVAL-1999-003132-05 (JCH):

Section 10.4.10.5: states the following- "Each turbine side drain pot is provided with an extraction line drain valve. These valves are opened and closed manually with control-board-mounted switches. After a valve opens, it remains open until manually closed.

Revise Section 10.4.10.5 to remove extraction line turbine side drain pot level switches and their associated level indication, level control and level alarm functions. Extraction Steam system drip pots (located between the Turbine and Feedwater Heater sides of the supply line stop and check valves) are not required to provide “Level Indication and Control Functions” in order to satisfy any CPSES commitments or applicable codes and standards. This change resolves the unit difference between Unit 1 and Unit 2.

LDCR-SA-2005-30, EVAL-2003-002540-04 (JCH):

Section 10.4.2: To ensure more reliable operation of the vacuum priming pumps year-round, a modification was implemented to provide full required seal water flow rate of 3 gpm per pump from the Demineralized and Reactor Makeup Water System. The water is not cooled and recirculated but is sent to the nearby floor drain as it exits the pump.

Removal of the vacuum primary pump heat exchangers ensures that the setpoint vacuum can be produced by the vacuum priming pumps. Other clarifications were to accurately reflect the system design per FDA-2003-002540-03 and -04.

LDCR-SA-2006-26, EVAL-2004-000897-02 (GLM):

In Section 10.4.12.5.1: Revise the first and second sentence to delete reference to the pitot tube type flow meter.

FDA-2004-00897-01 changes the way the TPCW total flow is determined for Unit 2. The existing Unit 2 total flow indication is erratic and varies depending upon which pump is running. To obtain a more reliable flow indication, the East, West and recirc flows are summed then transmitted to the control room. The control room flow indication and alarms will not be changed. Unit 1 is not modified (still has the pitot tube type flow meter), this change only removes the level of detail regarding the specific type of flow meter.

LDCR-SA-2005-17, EVAL-2004-002063-7 (JDS):

Revised Table 7.5.7E, sheet 4 of 4 to incorporate commitment made during license submittal for TSTF 447 for H2 recombiner removal. This was approved by the NRC in License Amendment 117.

LDCR-SA-2005-24, EVAL-2002-001952-3 (JDS):

Revised Ch. 1A(B) to provide discussion of deviations to Regulatory Guide 1.82 that are being addressed as part of resolution of the generic issue regarding containment emergency sump performance (GSI-191).

LDCR-SA-2005-24, EVAL-2002-001952-3 (JDS) (continued):

Provide for description of the transport of debris through the radiation protection doors as a result of modifications to improve containment emergency sump performance.

LDCR-SA-2006-13, EVAL-2006-000212-02 (JDS):

Section 4.4.2.2.2 - The FSAR description of the calculation of the fuel assembly spring holddown force has been revised. The use of the mechanical design RCS flow has been replaced by the use of the best-estimate RCS flow with margin.

The current Westinghouse methods involve the use of best estimate flow (vs mechanical design flow) to calculate best estimate lift forces. An uncertainty factor (calculated by convoluting uncertainties, including an uncertainty to account for the difference of best estimate flow to design flow) is then applied resulting in the design lift forces. These design lift forces are then used to ensure the functional compliance of the holddown springs. The hold down springs remain capable of performing their function.

Section 5.1, under Best Estimate Flow and Thermal Design Flow - The FSAR description of the calculation of the fuel assembly spring holddown force has been revised. The use of the mechanical design RCS flow has been replaced by the use of the best-estimate RCS flow with margin.

The current Westinghouse methods involve the use of best estimate flow (vs mechanical design flow) to calculate best estimate lift forces. An uncertainty factor (calculated by convoluting uncertainties, including an uncertainty to account for the difference of best estimate flow to design flow) is then applied resulting in the design lift forces. These design lift forces are then used to ensure the functional compliance of the holddown springs. The hold down springs remain capable of performing their function.

LDCR-SA-2006-33, EVAL-2001-001255-17 (TJE):

Section 8.3.1.1.11.1: Replace the words "Unit 1 and Unit 2 Diesel Generators use" with the words "Each EDG uses."

Currently, this paragraph reads,

"Unit 1 and Unit 2 Diesel Generators use an upgraded digital excitation/voltage regulator. This digital exciter/voltage regulator is a Safety Related Siemens THYRIPART generator field excitation unit. This THYRIPART excitation unit utilizes a digital control system consisting of a 32-bit technology module and communications module. This system controls the output of a chopper controller circuit to achieve +0.5 percent Diesel Generator voltage regulation, and has capability of both manual as well as automatic control."

LDCR-SA-2006-33, EVAL-2001-001255-17 (TJE) (continued):

The paragraph will be revised to read,

"Each EDG uses a digital excitation/voltage regulator. This digital exciter/voltage regulator is a Safety Related Siemens THYRIPART generator field excitation unit. This THYRIPART excitation unit utilizes a digital control system consisting of a 32-bit technology module and communications module. This system controls the output of a chopper controller circuit to achieve +0.5 percent Diesel Generator voltage regulation, and has capability of both manual as well as automatic control."

Section 8.3.1.1.11.2: 1.) Add the following item to the reason for the automatic tripping of the DGs

"t. Generator underfrequency"

Delete the following paragraph

"These protective functions on the Unit 1 and the Unit 2 DG units utilize an upgraded digital multifunction protective device (MPR-1) locally which generates a common alarm to the control room."

Delete the following paragraph,

"For Unit 1 and Unit 2, the individual protective alarms except for the differential protection signal target are available locally on device MPR-1."

LDCR-SA-2006-15, EVAL-2006-001324-02 (TJE):

Section 8.3.1.1.11.3: FSAR Section 8.3.1.1.11, sub-section 3. "Testability and Maintenance," is revised to remove portions of the FSAR description of Diesel Generator (DG) periodic testing that are no longer applicable. This section of the FSAR formerly included additional detail that distinguished between DG testing performed during unit at-power operation and that which was previously restricted to performance during unit shutdowns. Surveillance tests for the DGs and the Offsite AC Sources are performed in accordance with the requirements of Technical Specifications (TS) 3.8.1 and are further described in the corresponding TS Bases. This change updates the FSAR description of DG periodic testing to be consistent with the current TS 3.8.1 surveillance mode restrictions as revised by License Amendment 124, effective April 24, 2006.

FSAR Amendment 101

LDCR-SA-2006-18, EVAL-1999-003485-02 (NSH):

Changes to Table 10.4-18, Page 1 of 1, corrects the Heater Drain Pump discharge pressure from 200 psig to 600 psig, identify temperature as 400^{o}F and identify the split of the Equalization Header as a sample point. This corrects the current FSAR Table 10.4-18, “Sample Points Pressure and Temperature Conditions” table. The Table in the FSAR currently incorrectly states that the Heater Drain Pump discharge parameters are the same as the Heater Drain Tank Equalization Header.

LDCR-SA-2006-32, EVAL-2005-001011-02 (RJK):

In FSAR table 9.3-4 the RCS Pressurizer Steam Space minimum sample rate is 0.75 GPM; in COP-101A/B (section 5.4.1) the minimum sample rate is 0.6 GPM. Based on a calculation performed, the Reynolds number for flow at 0.75 GPM is 3869. The Reynolds number for flow at 0.6 GPM is 3096. While the Reynolds number is less that the value for 0.75 GPM the number is greater that laminar flow values, therefore the minimum flow of 0.6 GPM in COP-101A/B provides flow sufficiently above laminar values to provide a representative sample.

In FSAR table 9.3-4 the SI Accumulator Tank minimum sample rate is 0.75 GPM; in COP-202A/B (section 5.1) the minimum sample rate is 1 liter/min. Based on a calculation performed, the Reynolds number for flow at 0.75 GPM is 3869. The Reynolds number for flow at 1 liter/min is 1363. A search of several references indicates that a Reynolds number of <2000 - 2300 is considered laminar flow and a Reynolds number of >4000 indicate turbulent flow. Based on the calculation, the lower flow referenced in the COP is laminar in nature. To compensate for the lower flow rate the purge volume is increased to 30 liters to ensure a representative sample of the accumulator is collected for analysis. The only item sampled for in the SI accumulator is boron, the Technical Specification limits for boron is 2300-2600 ppm, boron at the specified concentration will remain soluble well below temperatures seen in containment. Also, no particulate sampling/analysis is required from the SI accumulators. Based on recent testing on the sample system while installing new throttle valves, 1 liter/min is the maximum flow rate that can be expected through the lines as constructed.

LDCR-SA-2006-5, EVAL-2002-004206-04 (RJK):

In FSAR Section 3.6B.2.5.2 the Unit 2 system design pressure has been established at 1860 psig to bound the maximum SI pump discharge pressure that could exist following a LOCA and while in the recirculation mode. The system piping and components have been re-evaluated for this higher design pressure. In Table 6.3-2 the relief valve set points have been changed to 1860 psig, i.e., just above the maximum postulated SI pump discharge pressure, to prevent their opening while in the recirculation mode. Increasing the relief valve set points from 1750 psig to 1860 reduces the probability that the valves will open while the SI system is operating in the recirculation mode and, therefore, provides greater assurance that they can perform their function as a boundary for a closed system outside containment.

LDCR-SA-2006-5, EVAL-2002-004206-04 (RJK) (continued):

The Unit 2 system design pressure has been established at 1860 psig to bound the maximum SI pump discharge pressure that could exist following a LOCA and while in the recirculation mode. The system piping and components have been re-evaluated for this higher design pressure.

In Table 6.3-1 the Unit 2 relief valve set points have been changed to 1860 psig, i.e., just above the maximum postulated SI pump discharge pressure, to prevent their opening while in the recirculation mode. Increasing the relief valve set points from 1750 psig to 1860 reduces the probability that the valves will open while the SI system is operating in the recirculation mode and, therefore, provides greater assurance that they can perform their function as a boundary for a closed system outside containment.

LDCR-SA-2005-26, EVAL-2005-002515-02 (JDS):

Revised sections 4.2, 4.3.2, 4.4.1, Appendix 4A Tables 4A.1.2-1, 4A.2.2-2, and 1.6-1: The Unit 1 Cycle 12 reload core design consists of 88 Westinghouse IFM/OFA fresh fuel assemblies, 89 Westinghouse Revised P+ once-burned fuel assemblies from U1C11(including one reconstituted with 3 stainless steel rods) and 16 twice-burned Westinghouse Revised P+ assemblies from U1C11. The Westinghouse Revised P+ design includes a small hole, debris-filtering bottom nozzle design, longer solid end plugs on the fuel rods, and an alternate "P-grid" located just above the bottom nozzle to trap any debris making it through the bottom nozzle. In addition to the above features, the U1C12 fresh IFM/OFA fuel assemblies include IFM grids for enhanced fuel performance and an oxide coating at the bottom 7 inches of each fuel rod for fuel rod protection. Oxide forms naturally on the fuel cladding during normal operation so this coating has no impact on fuel design function. The pressure drop through the Westinghouse Revised P+ assemblies is slightly lower than that through the co-resident fresh Westinghouse IFM/ OFA assemblies while the heat transfer characteristics are improved by the IFMs. The effects of the mixed core on the thermal-hydraulic response are considered. The impact of the stainless steel, unheated rods in the reconstituted assembly on the mechanical, nuclear and thermal-hydraulic designs was also considered.

Added COLR references from Section 5.6.5 of the Technical Specifications for completeness.

Added Zirlo reference for the specific cladding used by Westinghouse fuel.

Added discussion on ZIROL fuel, reconstitutided fuel, and Westinghouse Optimized fuel with axial blankets or integral fuel burnable absorber. These fuel types and options are loaded into Unit 1 Cycle 12.

Added fuel cladding discussions on fretting, cycling and fatigue, and irradiation stability of the cladding and reference.

Added Zirlo reference for the specific cladding used by Westinghouse fuel.

LDCR-SA-2005-26, EVAL-2005-002515-02 (JDS) (continued):

Added FRA-ANP (framatome) to cover both fuel vendors used.

Added references for to Westinghouse topicals (WCAPs) used to support Unit 1 cycle 12.

Add discussion for reconstituted fuel used in Unit 1 cycle 12.

LDCR-SA-2006-27, EVAL-2006-002148-02 (JCH):

Replace "Engineering Programs Manager" with "Technical Support Manager in paragrahs 13.3B.1.2 and 13.3B.1.4.

Move item 2 under paragraph 13.3B.1.4 to item 3 under paragraph 13.3B.1.2 and renumber subsequent items under 13.3B.1.2 and 13.3B.1.4.

Reassignment of responsibilities is due to reorganization as a result of business process outsourcing.

LDCR-SA-2006-29, EVAL-2001-002001-01 (DWS):

Table 17A-1: Allow permanent installation and erection of Scaffold poles inside Containment Steam Generators Compartments or Loop Rooms and permit storage racks for scaffold material outside the loop rooms at the lower elevation. Reference FDA 2006-002001-01.

LDCR-SA-2006-28, EVAL-2006-001288-03 (JDS):

Update of FSAR for the Unit 2 Cycle 10 core configuration and operation. This update replaces the current FSAR Section 4B in its entirety.

Add reference to section 9.1.4.2.1 for a description of the soluble boron credit.

Add additional description for the soluble boron credit. 9.1.4.2.1 describes the soluble boron credit

Add Step 5 for conservative actions to be performed for operator actions to be performed for primary to secondary break.

LDCR-SA-2007-1, EVAL-2006-002939-02 (DWS):

Section 13.4-1: Change organizational name from "Nuclear Overview" to "Quality Assurance".

Section 14.2.2.4.6 Change title from "Director of Nuclear Overview" to "Director, Oversight & Regulatory Affairs". Section 14.2.2.6 & 14.2.2.8 change organizational name from "Nuclear Overview" to Quality Assurance".

LDCR-SA-2007-1, EVAL-2006-002939-02 (DWS) (continued):

Section 1.B.1.2 CPSES Response - change organizational name from "Nuclear Overview" to "Quality Assurance".

Section 1.C.5 change organizational name from "Nuclear Overview Department" to "Performance Improvement Department" in three places.

Section 17.2: TOC Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs"

Section 17.2.1.1.2 Add additional responsibility for to the Site Vice President. Responsible for "Technical and administrative direction of the Director, Performance Improvement."

Section 17.2.1.1.3 Add responsibility for Plant Manager. Responsible for Tecnical and administrative direction to the Director, Plant Support.

Section 17.2.1.1.4 Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs", change organizational name from "Nuclear Overview" to "Quality Assurance", and revise duties and responsibilities for the Director Nuclear Oversight & Regulatory Affairs and add duties and responsibilities for the Quality Assurance Manager.

Section 17.2.1.2 Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs", change organizational name from "Nuclear Overview" to "Quality Assurance", and remove function number 10 "Administering and facilitating the corrective action program."

Section 17.2.1.5.1 Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs."

Section 17.2.2 Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs", change organizational name from "Nuclear Overview" to "Quality Assurance", add duties for the Quality Assurance Manager, remove evaluations and replace with "surveillance and audits", and remove the responsibility for "Administering and facilitating the corrective action program."

Section 17.2.5 Change organizational name from "Nuclear Overview" to "Quality Assurance", remove evaluations and replace with "surveillance and audits"

Section 17.2.6.1 Change title from "Director Nuclear Overview" to "Quality Assurance Manager", change organizational name from "Nuclear Overview" to "Quality Assurance"

Section 17.2.7 Change organizational name from "Nuclear Overview" to "Quality Assurance"

Section 17.2.9 Change "procedures shall be reviewed by the Nuclear Overview Department or designee" to "review by Quality Assurance or other qualified personnel."

LDCR-SA-2007-1, EVAL-2006-002939-02 (DWS) (continued):

Section 17.2.10 Change title from "Director Nuclear Overview" to "Quality Assurance Manager", change organizational name from "Nuclear Overview" to "Quality Assurance"

Section 17.2.12 Change organizational name from "Nuclear Overview" to "Quality Assurance" and add "TXU Power" in front of manager of the oraganization responsible for the calibration.

Section 17.2.14 & 17.2.15 Change organizational name from “Nuclear Overview” to “Quality Assurance”.

Section 17.2.16 Add "The Director, Performance Improvement is responsible for administrating and faciltating the corrective action program".

Section 17.2.18 & .1 & .2 & .2.1 Replace "evaluation" with audit, change title from "Director Nuclear Overview" to "Quality Assurance Manager", and change organizational name from "Nuclear Overview" to "Quality Assurance"

Section 13.1: TOC Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs", Change organizational name from "Nuclear Overview" to "Quality Assurance".

Section 13.1 Change organizational name from "Nuclear Engineering to Nuclear Engineering & Support; Regulatory Affairs to Oversight & Regulatory Affairs; Nuclear Overview to Quality Assurance.

Section 13.1.1.2.1 added "Senior" to title of "Vice President & Chief Nuclear Officer" and added position of "Director, Performance Improvement" with responsiblities outlined. Add "Supply Chain and Procurement to Director, Plant Support responsibilities. Removed position of "Emergency Planning Manager"

Section 13.1.1.2.2 Revise organization name to Nuclear Engineering & Support. Removed position of "Manager, Administrative Services" and moved responsibility to Manager, Project Engineering.

Section 13.1.1.2.3 Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs" and revise organizational name.

Section 13.1.1.2.4 Change organizational name from "Nuclear Overview" to "Quality Assurance", Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs", remove responsibilities for "Providing for the review and assessment of reports of nuclear industry operating experience.", "Administering and facilitating the corrective action program.", and "Provide for the trending and analysis of conditions adverse to quality." The responsibilities are now under the Director, Performance Improvement who reports to the Site Vice President.

Section 13.1.2.1.2 Rewrite responsibilities of "Shift Operation Manager"

LDCR-SA-2007-1, EVAL-2006-002939-02 (DWS) (continued):

Section 13.1.2.1.3 Revise organizational manager titles.

Section 13.1.2.1.4 Editorial changes

Section 13.1.2.1.5 Remove the organization "Central Organization for Reliable Plant Systems (CORPS). This organization has been reassigned and renamed.

Section 13.1.2.1.6 Delete section number (number only).

Table 13.1: Revise titles, remove PLANT MOD TEAM MANAGER and insert Project Engineering Manager, Change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs".

Section 13.1A-1: Insert resumes and titles for new personnel and change title from "Director Nuclear Overview" to Director, Oversight & Regulatory Affairs". Remove old titles and resumes.

Section 13.3 Delete redundant and historical information on Fire Protection.

Section 13.3B Editorial changes to Fire Protection discription.

Section 13.3B.1.2 & .1.3 & .1.4 & .1.5 Revise management titles

Section 13.4-1: Change organizational name from "Nuclear Overview" to "Quality Assurance"

Sections 13.5-1, 5, 7: Correct name of company from TXU Energy to TXU Power and revise management titles.

Figure 13.1-2: Added Director, Performance Improvement and re-alignment of reporting respondibilities of the Corrective Action Program Manager and the Quality Assurance Manager.

Figure 13.1-3: Added Director, Performance Improvement and change various management titles.

LDCR-SA-2004-42, EVAL-2004-000773-01 (JCH):

Revised sections 3.5.1, 5.4.1, 7.1.2, 7.3.2, 7.7.2, 9.5.1, 10.2.2, 10.2.3, 10.2.4 and Table 7.3-4: The requirement for a complete maintenance inspection of the mechanical overspeed protection system every refueling outage has been removed, based on the fact that the mechanical overspeed system has been removed and replaced with a digital electronic protection system. The new digital overspeed protection system is auotmatically tested periodically during each shift and is verified operational every 2 weeks. The turbine trip block is also tested every 2 weeks

LDCR-SA-2004-42, EVAL-2004-000773-01 (JCH) (continued):

Deleted mechanical overspeed trip since the device is removed and added words to describe the new electronic overspeed systems

Revised "3. Turbine" of this section. This section lists equipment that cannot be tested at full power so as not to damage equipment or upset plant operation. The turbine trip solenoids were tested at power every 2 weeks under OPT-217, "Protective devices test". Under the new system, the turbine trip block trip solenoid valves are testable online and will also be tested every 14 days. The Turbine Trip Block will be tested during each startup from refueling outage.

Inserted language to indicate that the slave-relay test results may be observed by the Operator on the OM screen

Remove the statement "A load demand greater than full power is prohibited by the turbine control load limit devices."

There were two such considered ‘load limiting devices.’ One was built into the original EHC control system. The operator could set this to limit the adjusted setpoint. This feature presented certain control problems that resulted in operator confusion and therefore not used. This feature was deleted when the TXP EHC control system was installed because there was no added value.

The second load limiting device is built into the mechanical hydraulic controller. The load limiting portion of the startup and load limiting device mechanically limited the hydraulic governor output which limited the hydraulic output of the electrohydraulic converter. This feature could limit speed since it acting on the hydraulic speed controller but could only limit megawatt output by limiting the amount the control valves could open due to the hydraulic limitation. This feature could not be effectively used to limit full output power because there is no accurate way to set it to perform this function and not interfere with the EHC control system.

The MHC is removed, and therefore, no longer available for this function. In reality, the MHC could never perform this function.

The only limitation to load is the normal load setpoint set by the operator on the operator control screen.

The following component are being removed and replaced by the new system:

- Mechanical Hydraulic Controller (MHC): The new system has enough built-in redundancy and diversity to protect the turbine against an overspeed event.The MHC is no longer needed with the new EHC control system which incorporates a new EH converter design.

Added fire test standard IEC60332-3.

LDCR-SA-2004-42, EVAL-2004-000773-01 (JCH) (continued):

Various pages in Section 10.2

The mechanical turbine trip devices are being deleted and replaced with a fully electronic trip system. The existing electronic turbine protection systems are being upgraded to a modern digital fail-safe system. Changes to the FSAR are necessary to remove references to deleted equipment as well as add section to describe the new system.

The new turbine protection system consists of a redundant automation system with redundant CPUs and a triple redundant fail-safe turbine trip system. The interface between the turbine trip system and the turbine hydraulics is performed by a redundant hydraulic trip block containing 3 trip solenoids. The turbine trip is performed with 2 of 3 trip logic resulting in rapid de-pressurization of the turbine trip fluid. The loss of trip fluid causes all stop and control valves to rapidly close isolating steam from the main turbine.

All of the analog trip signals are connected to the automation system. The redundant input signals are independent from each other by connecting them to different input/output modules with separate power supplies. The software with the applicable logic (2 of 2, or 2 of 3) is running on the redundant CPUs of the automation system. The automation system provides 3 redundant binary outputs (trip) signals to the fail-safe turbine trip system.

Binary inputs from field devices are directly connected to the fail-safe turbine trip system. Each binary input is connected to each channel of the turbine trip system. The output of each channel of the turbine trip system is connected to each normally energized trip solenoid (3 each) of the hydraulic trip block.

The turbine trip block is the interface between the turbine trip system and the turbine hydraulic system. The three trip solenoids of the trip block are normally energized by the turbine trip system. If any two trip solenoids valves are de-energized, the turbine will trip. This fail safe arrangement provides high degree of availability, reliability and protection of the turbine generator system.

The actual position of the trip piston in each individual channel of the trip block is measured by a position transducer. The position signals are used for status indication and feedback signals.

With the exception of the hydraulic trip devices, the new Automatic Turbine Tester (ATT) performs the same functions as the old ATT, however, the new design is implemented in software rather than hardware. The new ATT Trip Block test will be an operator initiated test.

LDCR-SA-2004-42, EVAL-2004-000773-01 (JCH) (continued):

Specific Changes

The following components are being removed and replaced by the new system:

a) Mechanical Hydraulic Controller (MHC): The new system has enough built-in redundancy and diversity to protect the turbine against an overspeed event.The MHC is no longer needed with the new EHC control system which incorporates a new EH converter design.

b) Load-limiting device: The load limiting device existed but was never used in the old system. Under the old system, the load limit was set at the maximum allowable, 1500 MWe, so that it wouldn't interfere with slight fluctuations that occurred during normal turbine operation. The new system does not have a load limit device. The MHC also had a so called load limiting device also. This was also no used for the same reason. The reference to load limit devices is being removed from the FSAR.

d) Extended Turbine Protection, which housed the automatic vibration trips and electrical low vacuum system. The electrical low vacuum system is incorporated into the new Turbine Protection system, where two out of three trip logic is performed in the software. The vibration trips will no longer be automatic, based on the following:

1) There is no code, standard, or regulatory requirement for these vibration trips.

2) The trips are not required by the plant insurer.

3) Operating experience with large turbine generators (fossil and nuclear) indicates that half of the turbine trips due to this system are spurious, and a vast majority of the remaining trips are also covered by other tripping events.

4) Several large turbine generator manufacturers (including Westinghouse) do not ship their equipment with this trip installed.

5) The high vibration alarm on the control board is retained, which will annunciate whenever any vibration is high to alert the operator. The operator will take the necessary action including turbine trip, if required to protect the turbine.

LDCR-SA-2005-16, EVAL-2004-001338-02 (JCH):

Revise section 3.9B: Delete the following: "After completion of testing , snubbers are disassembled and examined."

Snubbers are not disassembled after being tested, they are installed back in their locations unless they fail. If a snubber fails, then it may be disassembled to determine the cause of the failure.This is consistent with the ASME OM Code 1998 Edition through 2000 addenda and STA-742, "Snubber Surveillance Program."

FSAR Amendment 101, Supplement a

LDCR-SA-2006-2, EVAL-2004-001966-02 (TJE):

Justification: The new Seismic Monitoring System with only free-field ground motion input will not compromise the ability of the system to determine whether or not the OBE was exceeded and subsequent shutdown of both units per Appendix A of 10CFR100. Therefore, the new Seismic Monitoring System does not present any added risk to public health or nuclear safety.

Replace the discussion of RG 1.12 in 1A(B) that says,

"The installation of instrumentation for earthquakes in the CPSES plant is in conformance with the requirements of Revision 1 (4/74) of this regulatory guide."

with the words,

"The installation of instrumentation for earthquakes in the CPSES plant meets the intent of Revision 1 (4/74) of this regulatory guide with respect to the ability to determining exceedance of the OBE in a timely manner. Seismic instrumentation includes only the free-field triaxial accelerometer installed in the Yard. Determination of OBE exceedance will be based on the methods of ANSI/ANS-2.10-2003, "Criteria for the Handling and Initial Evaluation of Records from Nuclear Power Plant Seismic Instrumentation." "

1.) Delete all of section 3.7B.4.1 and insert the following:

"The seismic monitoring system meets the intent of RG 1.12, Rev. 1, with respect to the ability to determining exceedance of the OBE in a timely manner. Seismic instrumentation includes only the free-field triaxial accelerometer installed in the Yard. For the purpose of determining whether or not the OBE has been exceeded, ANSI/ANS-2.10-2003, "Criteria for the Handling and Initial Evaluation of Records from Nuclear Power Plant Seismic Instrumentation," uses the free-field accelerometer data as input to address 10CFR100 Appendix A requirements in paragraph (V)(a)(2). Exceedance above the OBE ground motion level will require structural response analyses to be performed (ANSI/ANS-2.23-2002, "Nuclear Power Plant Response to an Earthquake"). The seismic monitoring system used to record a seismic event is in accordance with the requirements described in ANSI/ANS-2.2-2002, "Earthquake Instrumentation Criteria for Nuclear Power Plants." The seismic monitoring system has the following components:

1. A triaxial accelerometer located in the free-field. The function of the triaxial accelerometer is to provide the acceleration time-history responses where the affects associated with surface features, buildings, and components will be insignificant.

2. A digital recorder that continuously monitors the free-field accelerometers with a solid state archival system. The function of the recorder is to provide a recording of the seismic event for post-event evaluation. The recorder will capture the 3-seconds preceding the exceedance of the seismic trigger threshold, and will continue to record data for a minimum of 5-seconds beyond the last recorded exceedance of the seismic trigger threshold.

LDCR-SA-2006-2, EVAL-2004-001966-02 (TJE) (continued):

3. A controller that will evaluate the recorded data in a timely manner. The function of the controller is to determine whether the OBE has been exceeded.

4. An uninterrupted power supply battery backup. The function of the battery backup is to ensure a minimum of 25-minutes of recorded data will be collected.

5. The free-field accelerometers will also function as the seismic switch. The recorder will continuously monitor the free-field accelerometers. The seismic trigger is a threshold ground acceleration value designed to initiate the event recording process at a value significantly below that of the OBE.

A schematic diagram of the seismic monitoring system is presented on Figure 3.7B-54. The free-field triaxial accelerometer described previously is provided for CPSES Unit 1 as allowed by Section 4.5 of ANSI/ANS-2.2-2002. The seismic monitoring system conforms to the requirements of ANSI/ANS-2.2-2002, as described above, including calibration and channel checks."

2.) Delete all of section 3.7B.4.2 and insert the following words,

"The recorder, controller, computer screen display, printer, and uninterruptible power system are located in an instrument rack in the Control Room. In case of any seismic activity of sufficient intensity to activate the seismic monitoring system, the Control Room operator is alerted by means of the seismic annunciation system, which consists of visual and audible alarms. The first notification will be that an event is in progress by the exceedance of the seismic trigger threshold and activation of the recorder. The second notification will be provided by the controller to indicate that the OBE has been exceeded and that shut down is required. A redundant set of notification indications is provided on the front of the seismic monitoring system in the form of light-emitting diodes (LEDs).

Delete all of section 3.7B.4.3 and insert the following words:

"This equipment computes and displays the free-field response spectra automatically on the computer screen following the event and prints a hardcopy on the system's printer. No Operator action is required to compute and print the response spectra. The seismic monitoring instrumentation facilitates the Operators ability to rapidly determine whether or not the OBE has been exceeded. In the event that the system fails to provide annunciation to one or both Control Rooms, local LED indication is provided on the face of the controller. The Control Room Staff will be alerted to the seismic event through perception of the building motions and external news sources. In addition, the control room operator is provided various screen displays that present the results of the analysis of the ground motion data. Therefore, failure to fully annunciate will not prevent the Control Room Staff from being alerted to the situation. The operator is required to request maintenance and engineering support to evaluate the validity of any alarm or indication of a seismic event. In response to a seismic event, the operator is required to perform focused walkdowns to assess plant damage and the availability of safe shutdown

LDCR-SA-2006-2, EVAL-2004-001966-02 (TJE) (continued):

equipment. The operator is required to shutdown both units if the Seismic Monitoring System determines that the OBE has been exceeded. In the event that the System is not available or functioning, the operator is required to request engineering to determine whether or not the OBE was exceeded."

Delete all of section 3.7B.4.4 and insert the following words,

"Recorded actual time histories from a significant seismic event that has occurred at the site are used to determine whether the OBE has been exceeded. The acceleration response spectrum from the actual time histories will be compared to the design spectra in the FSAR Figures 3.7B-1 and 3.7B-6. Due to large conservatism in the generation of in-structure responses, a detailed comparison of measured to predicted responses is not required."

This figure will be revised to reflect the new seismic instrumentation schematic diagram. See EVAL-2004-001966-01 for the markup.

LDCR-SA-2007-4, EVAL-2006-003263-04 (TJE):

Delete the "F" in front of the word "DISCUSSION" from the last entry in the Table of Contents

1A(B) FDISCUSSION OF REGULATORY GUIDES

In 1A(B), under the discussion of exceptions to compliance with RG 1.137:

1.) add "[49]" after "ANSI N195-1876" in item 1

2.) in item 4.a, add a "*" before and "when tested in accordance with ASTM D1298-1980" after the words "API or Specific Gravity"

3.) in item 4.d, remove the sapce between "D4176-" and "1982" and replace "ASTM-D1796-1968" with "ASTM D1796-1986"

4.) In the paragraph after item 4.d, near the end of the sentence, replace the "or" with "," and add ", or ASTM D4294-2003" at the end of the sentence.

5.) Add the following note after the paragraph that follows section 4.d,

"*If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of "greater than or equal to" 0.8348 or for API gravity at 60°F of "less than or equal to" 38°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations."

LDCR-SA-2007-4, EVAL-2006-003263-04 (TJE) (continued):

At the end of the discussion of exceptions to RG 1.137, add exception 9:

"9. C.2.a

If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of "greater than or equal to" 0.8348 or for API gravity at 60°F of "less that or equal to" 38°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations."

At the end of the Reference secion in 9.5, add reference number 49,

"49. ANSI N195-1976, 'Fuel Oil Systems for Standby Diesel Generators'"

In the paragraph before section 9.5.4.2.2, add the following paragraph,

"If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of "greater than or equal to" 0.8348 or for API gravity at 60°F of "less than or equal to" 38°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations."

Before section 9.5.4.2.2.6, add the following paragraph,

"If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of "greater than or equal to" 0.8348 or for API gravity at 60°F of "less than or equal to" 38°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations."

After the second paragraph of section 9.5.4.4, add the following paragraph,

"If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of "greater than or equal to" 0.8348 or for API gravity at 60°F of "less than or equal to" 38°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations."

LDCR-SA-2007-3, EVAL-2002-000162-06 (GLM):

Revise section 9.3.4.1.1.1 to indicate that if the BTRS chiller and associated components are used in the future that the flow and piping vibration response will be verified prior to use.

This change clarifies the description of the BTRS to add the commitment for testing the BTRS chiller and associated components. Since load follow ops are not used at CPSES,

LDCR-SA-2007-3, EVAL-2002-000162-06 (GLM) (continued):

there are no current plans to ever use these components. The addition of this commitment to perform testing prior to use in section 9.3.4.1.1.1 ensures that this testing will be incorporated into operating procedures if this part of the system is ever used in the future.

LDCR-SA-2006-4, EVAL-2005-000657-01 (RJK):

This change reflects construction of a long-term storage facility (OSGSF) for the long-term storage of four Unit-1 Old Steam Generators, one Unit -1 Old Reactor Vessel Storage head and one Unit-2 Old Reactor Vessel Head. The design function of the OSGSF is to limit the on-site and off-site radiation exposure from the old SGs and old RVHs and any attached components that are stored inside the OSGSF. The OSGSF provides sufficient shielding to meet the external dose limits specified in 10CFR20 and 40CFR190. Any postulated failure of the OSGs and ORVHs/CRDMs, due to the collapse of the OSGSF due to seismic event or any other natural phenomenon, or due to the impact of the components with the ground when they are dropped, will not result in an off-site dose greater than 0.5 rem to the whole body or its equivalent to any part of the body in accordance with the 10CFR20 and 40CFR190.

This change reflects construction of a long-term storage facility (OSGSF) for the long-term storage of four Unit-1 Old Steam Generators, one Unit -1 Old Reactor Vessel Storage head and one Unit-2 Old Reactor Vessel Head. This change extends the area shown on the site plot to now include that new facility.

LDCR-SA-2005-12, EVAL-2005-001957-02 (CBC):

FSAR Sections 1, 6 and 15 are revised to reflect new methods an assumptions for evaluating radiological consequences for design basis accidents which are adopted consistent with NRC Regulatory Guide 1.195 and the new Control Room Habitablity Program / Control Room Integrity Program. These methods and assumptions were approved by the NRC in License Amendment 130. See TXX-06151 and TXX-06193 for additional information.

LDCR-SA-2007-12, EVAL-2004-002120-12 (TJE):

LDCR SA-2007-012 removes any ambiguity from the FSAR that the spare Startup Transformer XST1/2 can be energizied promptly, via a motor operated switch, is available to back up XST2.

1.) In the first full paragraph on page 8.2-2, replace the word "The" with "A" such that the sentence reads:

"A spare startup transformer, XST1/2 with dual primary windings (345-kV and 138-kV), is stored in a dedicated location under the 345-kV line to XST2 (refer to Figure 8.2-1)."

LDCR-SA-2007-12, EVAL-2004-002120-12 (TJE) (continued):

2.) In the next sentence of the same paragraph, replace the words "can be used" to the words "must be physically relocated", such that the sentence reads,

"This transformer must be physically relocated to replace XST2 or XST1 if required."

LDCR-SA-2006-42, EVAL-2003-002426-21 (GLM):

Revise Figure 1.2-1 to show deletion of the RCA Yard Entrance Radiation Protection Building.

The RCA Yard Entrance Radiation Protection Building was removed during the Unit 1 Steam Generator Replacement Project. The building has no radiation protection functions and only functions as storage and an access point into the Unit 1 yard area.

Add "Item 8" to Section 3.8.1.1.6 which is titled "Containment Alternate Access for the Steam Generator and Reactor Pressure Vessel Head Replacement (Unit 1)." Item 8 details the codes and specification used for the Unit 1 RSG Containment Alternate Access (CAA) restoration.

Add reference to structural welding code AWS D1.4-1998 in Section 3.8.1.2.4.

This code was used for the Unit 1 Containment Alternate Access during the Steam Generator Replacement Project.

Section 3.8.1.6.2, Item 1, add reference to ASTM A706 for Unit 1. Section 3.8.1.6.2, Item 3, add references to codes and standards used in support of the Unit 1 Steam Generator Replacement Project.

These codes and standards were used in support of the Unit 1 Steam Generator Replacement Project.

Clarification to sections 3.8.1.6.3 and 3.8.1.6.4 related to codes and standards used in support of the Unit 1 Steam Generator Replacement Project.

These codes and standards were used in support of the Unit 1 Steam Generator Replacement Project.

LDCR-SA-2006-40, EVAL-2003-002426-20 (RJK):

Update reference power in Section 1.2.2 for Unit 1 to 3458 MWt to reflect uprate that was implemented in 1RF09.

Justification: this 1.4% power uprate previously occurred during 1RF09 as implementation of License Amendment 89..

Adds five new references to Section 1.6 to reflect RSGs.

LDCR-SA-2006-40, EVAL-2003-002426-20 (RJK) (continued):

Adds new code reference in Section 1A(N) for Unit 1 RSGs (ANSI N45.2.2-1978) in RG 1.38 discussion.

Updates material descriptions in RG 1.43 discussion of Section 1A(N) to reflect RSGs and new RX head.

Adds new code reference in Section 1A(N) for Unit 1 RSGs (ANSI/ASME NQA-1) in RG 1.64 discussion.

Updates Table 1A(N)-1 to reflect Unit 2 only for RG 1.66 applicability.

Updates accident analysis discussions in 3.6B to specify new analyses for Unit 1 RSGs.

Updates text references in 3.6B to identify separate Tables for Unit 1 and Unit 2 parameters.

Adds 8 new references to Section 3.6B to reflect RSGs.

Split Table 3.6B-4 into separate Tables (4A and 4B), one for each unit.

Added NOTE that Table 3.6B-6 is for original SGs only and that Unit 1 RSGs had separate mass and energy release data calculated with RELAP5/MOD3.

Specified that discussion of critical damping values in Section 3.7N.1.3 are for Unit 2 only. Unit 1 is now discussed in Section 3.9N.1.4.3.

add the following statement to Sections 3.8.4.3.2 and 3.8.4.3.3:

"See Section 3.6B.1.2.3 "Structural Design Margins" for additional requirements in the MS/FW penetration areas inside the Safeguards Building."

Update Section 3.9N listing of Computer Programs Used in Analyses to reflect RSG models.

Update Section 3.9N discussion of seismic design considerations in Sections 3.9N.1.4.2, 3.9N.1.4.3, 3.9N.1.4.4, and 3.9N.2.5 to reflect RSGs.

Add eight new references to reflect RSG models to Section 3.9N.

Add the following NOTE to Figure 3.9N-1:

"FOR UNIT 1, THE REPLACEMENT STEAM GENERATOR ACTUALLY HAS 8 MASSES DISTRIBUTED ALONG THE VERTICAL CENTERLINE, AND THE SNUBBERS HAVE BEEN ELIMINATED FROM THE UPPER SUPPORT"

Split Table 3.9N-14 into separate Tables (14A and 14B), one for each unit.

LDCR-SA-2006-40, EVAL-2003-002426-20 (RJK) (continued):

Split Table 3.9N-15 into separate Tables (15A and 15B), one for each unit.

Split Table 3.9N-16 into separate Tables (16A and 16B), one for each unit.

Split Table 3.9N-21 into separate Tables (21A and 21B), one for each unit.

Split Figure 4.4-21 into separate Figures (21A and 21B), one for each unit.

Split Table 5.1-1 into separate Tables (1A and 1B), one for each unit.

Added applicable code cases used for manufacture of RSGs; N-20-4, 2124-1, and 2143-1 to Section 5.2.

Updated Table 5.2-1 to reflect new code addenda for Unit 1 RSGs and RX head.

Updated Table 5.2-2 to reflect new RCS pressure boundary materials for Unit 1 RSGs and RX head.

Split Table 5.2-6 into separate Tables (6A and 6B), one for each unit.

Updated Table 5.2-7 to reflect applicable code cases used for manufacture of RSGs; N-20-4, 2124-1, and 2143-1.

Replace the data in the first 8 lines of Table 5.3-16A to reflect the Fracture Toughness Test report numbers for the RSGs.

Add weld metal fracture toughness data for Unit 1 Replacement Steam Generator in Table 5.3-17A.

Add new section 5.4.2A to discuss Unit 1 RSG materials. Renumber existing Section 5.4.2 to 5.4.2B and make it Unit 2 specific.

Update hotlinks throughout section to either 5.4.2A or 5.4.2B as appropriate.

Update references in Section 5.4.2 to reflect Unit 1 RSGs.

Update SG design data in Table 5.4-3 to reflect Unit differences between Unit 2 original SGs and Unit 1 RSGs.

Split Table 5.4-4 into separate Tables (4A and 4B), one for each unit.

Split Figure 5.4-4 into separate Figures (4A and 4B), one for each unit.

Add the following NOTE to Figure 5.4-13:

"The snubbers exist only on Unit 2. They have been removed from Unit 1."

LDCR-SA-2006-40, EVAL-2003-002426-20 (RJK) (continued):

Updated discussion of Compliance with BTP RSB 5-1 (Diablo Canyon RHR comparison) in Section 5A to reflect separate calculations for each unit.

Update chart in Section 12.1 to reflect differing cobalt Wgt% for Unit 1 RSG tubes.

Update discussion in Section 12.3 of SG tubing materials to reflect RSGs and material selection.

Update Section 1.2.2 nominal performance characteristics of NSSS to reflect RSGs in Unit 1. Includes addition of two new reference documents into new section 1.2.2.13.

Justification:

- Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2005-000659-02, FDA-2005-000659-03 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

editorial change on page 5.3-4 to correct typographic error

Justification:

- Typographic error only, no change in technical content.

LDCR-SA-2006-37, EVAL-2003-002426-17 (JCH):

Insert the following into Section 10.3.2.2, "Atmospheric Relief Valves":

In Unit 1 the Westinghouse steam generators model D-4 have been replaced with a newer model D-76. Two solenoid valves are installed for each ARV. These solenoid valves are supplied from the opposite train power supply. These solenoid valves are operated from a key locked hand switch in the main control room to open the ARVs.

During a Unit 1 steam generator fault, if normal DC power is lost to the Train A or B buses, the ARVs can still be opened if required from the opposite train power to cool and depressurize the reactor and terminate primary to secondary leak.

Justification: The replacement steam generators (RSGs) installed in Unit 1 are physically different than the original SGs and when the SGTR analysis was revised to reflect the thermal hydraulic performance of the RSGs under the accident conditions, one ARV was not sufficient for the primary side cooldown and additional relief capacity is required.

In Section 10.4.8.1.1, change the total combined blowdown flow for Unit 1 from "69, 600" to "155,620".

LDCR-SA-2006-37, EVAL-2003-002426-17 (JCH) (continued):

Justification: RSG have higher SGBD flow.

Added "balancing" before "valves HV-5175 through HV-5178".

Add the following sentence:

"Once the balancing valves are set they are locked in position and do not change.

Justification: clarification on how to set blowdown for each steam generator.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH):

DESCRIPTION: Appendix 3B

Page 3B-ii: For computer program 44 delete "*" before "ME215" and insert "(ME101FE)" after "ME215.

Page 3B-iv: Section 3.44 delete "*" before "ME215" and insert "(ME101FE)" after "ME215.

Page 3B-54: Insert "(ME101FE)" after "ME215" in the section title and in the first paragraph and delete "(Unit 2 only)" in the section title.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01.

These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION: Section 3.6B

3.6B.1.2.3.E - Break Exclusion Areas (Feedwater)

Page 3.6B-7 second paragraph last sentence after "..... region" insert "(Unit 2 only)" since the feedwater bypass piping has been eliminated for the Unit 1 RSG.

3.6B.2.2.2 - High - Energy Piping Other Than RCS Main Loop

Page 3.6B-22 second paragraph first sentence insert "or 34" after "[6" for new Reference 34 and in the last sentence after "[29]" insert "or other" and delete the comma and after "programs to" insert "the."

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued):

Page 3.6B-23 first paragraph first sentence after "[30" insert "or 35" for new Reference 35.

Page 3.6B-53 add new References 34 and 35. Reference 34 is "RELAP5/MOD 3.2, NUREG/CR-5535, June 1995" and Reference 35 is "ANSYS, Version 8.1."

3.6B.2.5.2.2 - Feedwater System - A. General Description

Page 3.6B-40 after first sentence insert "Due to the Unit 1 RSG installation the 18" x 16" reducing elbow material at the RSG nozzle inlet is now SA-508, Grade 2, Class 1."

3.6B.2.5.2.3 - Auxiliary Feedwater System - D. Environmental Analysis

Page 3.6B-43 after "........250F maximum" insert "(Unit 2 only)."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

For convenience the main steam system, steam generator blowdown system and miscellaneous FSAR figures pertaining to stress node break point and restraint location have been included in this EVAL.

Fig 3.6B-11 - Main Steam System: Loop 1 Inside Containment Stress Node Break Point and Restraint Location

Changed break 8C stress point to 22 and break 1C stress point to SGB and revised the legend.

Fig 3.6B-12 - Main Steam System: Loop 2 Inside Containment Stress Node Break Point and Restraint Location

Changed break 13C stress point to 28 and break 9C stress point to SGB and revised the legend.

Fig 3.6B-13 - Main Steam System: Loop 3 Inside Containment Stress Node Break Point and Restraint Location

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Changed break 14C stress point to N03 and revised the legend.

Fig 3.6B-14 - Main Steam System: Loop 4 Inside Containment Stress Node Break Point and Restraint Location

Changed break 29C stress point to 800 and revised the legend.

Fig 3.6B-19 - Feedwater System: Loop 1 Inside Containment Stress Node Break Point and Restraint Location

Changed break 401C stress point to D04 and break 410C stress point to V09. Revised the notes and legend. Added snubbers and spring hangers to the drawing and deleted split flow bypass line. Revised piping for elevated RSG FW nozzle and deleted chemical feed system piping. Deleted lines capped.

Fig 3.6B-20 - Feedwater System: Loop 2 Inside Containment Stress Node Break Point and Restraint Location

Changed break 411C stress point to 50 and break 418C stress point to B50. Revised notes and legend. Added snubbers, rigid restraints and spring hangers to the drawing and deleted split flow bypass line. Revised piping for elevated RSG FW nozzle and deleted chemical feed system piping. Deleted lines capped.

Fig 3.6B-21 - Feedwater System: 1-3 Inside Containment Stress Node Break Point and Restraint Location

Changed break 414C stress point to H01 and break 434C stress point to V09. Revised notes and legend. Added snubbers, rigid restraints and spring hangers to the drawing and deleted split flow bypass line. Revised piping for elevated RSG FW nozzle and deleted chemical feed system piping. Deleted lines capped.

Fig 3.6B-22 - Feedwater System: Loop 4 Inside Containment Stress Node Break Point and Restraint Location

Changed break 421C stress point to 35 and break 429C stress point to N02. Revised notes and legend. Added snubbers, rigid restraints and spring hangers to the drawing and deleted split flow bypass line. Revised piping for elevated RSG FW nozzle and deleted chemical feed system piping. Deleted lines capped.

Fig 3.6B-26 - Aux. F.W. System Stress Node Break Point and Restraint Location (Units 1 and 2)

Revised to show preheater bypass lines (6-FW-1-91-2003-2 through 6-FW-1-94-2003-2) are for Unit 2 only.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Fig 3.6B-27 - Auxiliary F.W. System: Loop 1 Outside Containment Stress Node Break Point and Restraint Location

Deleted preheater bypass line with a capped pipe section and revised the legend.

Fig 3.6B-28 - Auxiliary F.W. System Outside Containment Stress Node Break Point and Restraint Location

Deleted preheater bypass line with a capped pipe section and revised the legend.

Fig 3.6B-29 - Auxiliary F.W. System: Loop 3 Outside Containment Stress Node Break Point and Restraint Location

Deleted preheater bypass line with a capped pipe section and revised the legend.

Fig 3.6B-30 - Auxiliary F.W. System Outside Containment Stress Node Break Point and Restraint Location

Deleted preheater bypass line with a capped pipe section and changed break 575C stress point to 34J. Revised the legend.

Fig 3.6B-34 - Feedwater System: Loop 1 Inside Containment Stress Node Break Point and Restraint Location

Changed break 597C stress point to 30 and break 598C to stress point B63. Deleted break 599C. Deleted split flow bypass line. Revised piping to the RSG AFW nozzle. Indicated that the chemical feed system and drain piping 2-1303-2 are eliminated. Drain piping 2-1303-2 was previously eliminated. Revised the legend and deleted 1-152A from the title block.

Fig 3.6B-35 - Feedwater System: Loop 2 Inside Containment Stress Node Break Point and Restraint Location

Changed break 628C stress point to 77 and break 629C to stress point 152. Deleted break 630C. Deleted split flow bypass line. Revised piping to the RSG AFW nozzle. Deleted chemical feed system piping. Deleted drain piping 2-1303-2 (previously eliminated). Revised the legend.

Fig 3.6B-36 - Feedwater System: Loop 3 Inside Containment Stress Node Break Point and Restraint Location

Changed break 622C stress point to A10 and break 623C to stress point NZ3. Deleted break 624C. Deleted split flow bypass line. Revised piping to the RSG AFW nozzle. Deleted chemical feed system piping. Deleted drain piping 2-1303-2 (previously eliminated). Revised the legend and deleted 1-154A from the title block.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Fig 3.6B-37 - Feedwater System: Loop 4 Inside Containment Stress Node Break Point and Restraint Location

Changed break 575C stress point to 34J and break 577C to stress point N03. Deleted break 576C. Deleted split flow bypass line. Revised piping to the RSG AFW nozzle. Deleted chemical feed system piping. Deleted drain piping 2-1303-2 (previously eliminated). Revised the legend.

Fig 3.6B-38 - STM. GEN. Blowdown Sys. Loop 1 Inside Containment Stress Node Break Point and Restraint Location

Changed break 202C stress point to A80, break 203C stress point to W85 and break 201C to stress point Q68. All intermediate breaks are deleted (21C, 22C, 48C, 113C, 116C, 117C, 139C, 205C, 206C, 208C, 1101C, 1160C, 1600C, 2100C and 8100C). Revised piping from the RSG blowdown and supplemental blowdown connections. Revised the legend.

Fig 3.6B-39-1 - STM. GEN. Blowdown Sys. Loop 2 Inside Containment Stress Node Break Point and Restraint Location

Changed break 192C stress point to 260, break 185C stress point to E01 and break 184C to stress point B67. All intermediate breaks are deleted (200C, 1127C/L, 1903C and 9006C). Revised piping from the RSG blowdown and supplemental blowdown connections. Revised the legend.

Fig 3.6B-40 - STM. GEN. Blowdown System Loop 3 Inside Containment Stress Node Break Point and Restraint Location

Changed break 45C stress point to 545. Intermediate break 9624C/L is deleted. Revised piping from the RSG blowdown and supplemental blowdown connections. Revised the legend.

Fig 3.6B-41 - STM. GEN. Blowdown System Loop 4 Inside Containment Stress Node Break Point and Restraint Location

Revised piping from the RSG blowdown and supplemental blowdown connections. Deleted chemical feed system piping. Revised the legend.

Fig 3.6B-55 - Safety Injection System Inside Containment Stress Node Break Point and Restraint Location

Break 8TCA moved to downstream side of check valve 1SI-8819C.

Fig 3.6B-67-1 - Reactor Coolant System Inside Containment Stress Node Break Point and Restraint Location

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Breaks 71C, 72C, 73C, 80C, 81C, 82C, 83C, 96C, 140C and 142C where revised to 4370, 88, 89, 1105, 5210, 105, 106, 82, 102/1115 and 85/82 respectively.

Fig 3.6B-82-1 - CVCS Outside Containment Stress Node Break Point and Restraint Location

Deleted breaks 1421-1C and 2C based on piping analyses (CS-1-046A-PB, Rev. 4). Added pipe break analysis calculation (CS-1-042A-PB, Rev. 2) to provide the source for break 1324C inside the Reactor Building illustrated on Isometric 1-46A.

Figure 3.6B-208 - Safeguard Building Main Steam & Feedwater Lines Arrangement

Sheet 2 of 3 show that the preheater bypass lines 6-FW-1-091-2003-2 through 6-FW-1-094-2003-2 are for Unit 2 only.

Sheet 3 of 3 Sections "53-53" and "54-54" show that the preheater bypass lines 6-FW-1-091-2003-2 through 6-FW-1-094-2003-2 are for Unit 2 only.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION: Table 3.6B-1 - High Energy Line List (See Notes 22 and 23)

Page 11 of 31:

Revise "Line Number" "3FW-1-93-203-2" to "6FW-2-93-203-2" and under "Remarks" revise the note to be "(Notes 1 and 25)." Under "Remarks" for "Line Number" "6FW-1-97-1303-2" insert "(Note 26)" after "bypass line."

Top of page under "Remarks" revise note to "(Notes 1 and 26)" after "bypass." Revise "Line Number" "6FW-1-101-1303-2" to "6FW-2-101-1303-2" and under "Remarks" insert "(Note 25)" after "SG Number 3." Revise "Line Number" "6FW-1-92-2003-2" to "6FW-2-92-2003-2" and under "Remarks" revise note to "(Notes 1 and 25)" after "bypass line."

Page 12 of 31:

Under "Remarks" for "Line Number" "6FW-1-96-1303-2" revise note to "(Notes 1 and 26)" after "bypass line."

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Bottom of page after "bypass line" revise note to "(Notes 1 and 26)."

Revise "Line Number" "6FW-1-100-1303-2" to "6FW-2-100-1303-2" and under "Remarks" insert "(Note 25)" after "SG Number 2." Revise "Line Number" "6FW-1-91-2003-2" to "6FW-2-91-2003-2" and under "Remarks" revise note to "(Notes 1 and 25)" after "bypass line."

Page 13 of 31:

Under "Remarks" for "Line Number" "6FW-1-95-1303-2" revise note to "(Notes 1 and 26)" after "bypass line." Bottom of page after "bypass line" revise note to "(Notes 1 and 26)."

Revise "Line Number" "6FW-1-99-1303-2" to "6FW-2-99-1303-2" and under "Remarks" insert "(Note 25)" after "SG Number 1." Revise "Line Number" "6FW-1-94-2003-2" to "6FW-2-94-2003-2" and under "Remarks" insert "(Note 25)" after "bypass line."

Revise "Line Number" "6F2-1-93-1303-2" to "6FW-2-93-1303-2 and under "Remarks" revise note to "(Notes 1 and 26)."

Page 14 of 31:

Revise "Line Number" "6FW-1-102-1303-2" to "6FW-2-102-1303-2."

Top of page under "Remarks" insert "(Note 25)" after "SG Number 4."

Revise "Line Number" "2FW-1-900-1303-2" through "2FW-1-907-1303-2" to "2FW-2-900-1303-2" through "2FW-2-907-1303-2."

Page 31 of 31: Insert the following notes:

25. Unit 2 only. Line eliminated on Unit 1 with RSG installation.

26. Preheater Bypass Line Unit 2 only. For Unit 1, line partially removed from connection to 18" main feedwater line to connection with 4" auxiliary feedwater line. Remaining portion of line is for auxiliary feedwater flow only."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

DESCRIPTION:

3.9B.1.2.6 - SAPCAS (ME 215) Stress Analysis for Pipe Containment and Pipe Support Using Finite Element Method - Unit 2 only

Page 3.9B-9: In section title insert "(ME101FE)" after "(ME 215)" and delete "- Unit 2 only." ME 215 is now called ME101FE. Both Unit 1 and Unit 2 have been analyzed using ME215 (ME101FE).

First paragraph after "ME 215" insert "(ME101FE)."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 3.9B-1F - Uses of Code Case N-318

Revise pipe support "FW-1-017-717-S525" to "FW-1-017-717-C52S" to correct typographical error.

Pipe support FW-1-102-700-C62R deleted based on the elimination of the split flow bypass line on Loop 4.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 3.9B-10 - Active Valves

Page 5: Insert "2" before "FV-2193 through 2196."

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Pages 5 and 6: Insert "2" before "FW-0191 through 0194 and FV-2181 and -2182."

Page 6: Insert "2" before "FV-2183 through 2184."

The above mentioned valves are deleted on Unit 1 for the RSG installation.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Figure 6.2.4-1 - Containment Isolation Valving

Sheet 9 of 12, Valve Arrangement 36: Make the "feedwater preheater bypass line" "Unit 2 only" with the piping ending between the check valve and junction with the "auxiliary feedwater" line.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 6.2.4-2 (Sheet 3) Containment Isolation Valving Application (Note 8)

Item "20c" insert "2-" before "FV-2193 and delete "29'-6." Item "22c" insert "2-" before "FV-2194 and delete "29'-6." Item "24c" insert "2-" before "FV-2195 and delete "30'-9." Item "26c" insert "2-" before "FV-2196 and delete "31'-10.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 6.2.4-3 (Sheets 2 and 3 of 13) Containment Isolation Valving Application (Note 1)

Item "20c", "Remarks" insert after signal "(Unit 2 only, valve deleted Unit 1)." Item "22c", "Remarks" insert after signal "(Unit 2 only, valve deleted Unit 1)." Item "24c", "Remarks" insert after signal "(Unit 2 only, valve deleted Unit 1)." Item "26c", "Remarks" insert after signal "(Unit 2 only, valve deleted Unit 1)."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 6.2.4-6 (Sheets 3 and 4) Classification of Systems Paths Penetration Containment Wall

Item "20a" under "System" insert "/AFW" and item "20c" after "Steam Generator #1 insert "(Unit 2 only)." Item "22a" under "System" insert "/AFW" and item "22c" after "Steam Generator #2 insert "(Unit 2 only)." Item "24a" under "System" insert "/AFW" and item "24c" after "Steam Generator #3 insert "(Unit 2 only)." Item "26a" under "System" insert "/AFW" and item "26c" after "Steam Generator #4 insert "(Unit 2 only)."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

DESCRIPTION:

7.1.2.5 Conformance to Regulatory Guide 1.22

Page 7.1-18, items 9 and 11 insert "Unit 2" before "Feedwater."

Page 7.1-21, item 9, insert "(Unit 2 only)" after "(Close)."

Page 7.1-22, item 10 insert "Unit 2" before "FWIV" and "FIBVs" and item 11 insert "(Unit 2 only)" after "(close)."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

7.3.1.1.4 - BOP Furnished Engineered Safety Features Systems - (5. Auxiliary Feedwater System,

Pages 7.3-14 and 15: In paragraph starting with "The auto start...." insert "the Unit 2" before "feedwater split flow bypass valves." Under "d. Interlocks" insert "the Unit 2" before "feedwater splitflow bypass valves."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 7.3-4 Safety Injection Actuated Equipment List

Sheet 2 of 21: Delete lines beginning with IFV-2181, 2182, 2183, 2184, 2193, 2194, 2195 and 2196.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 7.5-5 Type D - Variables

Sheet 2 of 6: Insert "Unit 2" before "Feedwater Split-flow."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 7.5-7B instrument Summary Data for Accident Monitoring Variables for CPSES not in Table 2 of Reg. Guide 1.97, Rev. 2

Sheet 4 of 10: Insert "2-" in front of "FV-2193 to 2196" and "FV-2181 to 2184."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

DESCRIPTION:

Figure 7.7-14 Main Control Board Functional Layout Units 1 and 2

For CB-09 make the "Feedwater FW WTR. HMMR CONT" "Unit 2 only."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Table 10.3-6 - Materials of Main Steam and Feedwater Valves and Piping

Sheet 2 of 2: Under "Fittings", "3 in. to 24 in." insert a superscript "1" for Note 1 after "WPL-6" and insert the following after the "Flanges" section "Notes: 1. Due to the Unit 1 RSG installation the 18" x 16" reducing elbow material at the RSG nozzle inlet is now SA-508, Grade 2, Class 1."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

10.4.7.2. - 2. Feedwater System

Page 10.4-32: Sixth paragraph first sentence revise sentence to include "When originally constructed both Unit 1 and Unit 2 incorporated a Feedwater Bypass System on each main feedwater line to a steam generator."

Insert the following paragraph: "With the installation of Delta 76 steam generators on Unit 1, the Feedwater Bypass System is no longer required by the new feedring steam generators and with the exception of the small bypass line around the main feedwater isolation valve has been totally removed from Unit 1. (Note: The Unit 1 feedwater system is operated in a different manner after RSG installation. The FIV remains open during

startup and a small flow through the valve and not the FIBV is used for purging purposes.) Unit 2 however still retains the Feedwater Bypass System in its entirety."

Add "The Unit 2" after paragraphs discussing feedwater bypass system, feedwater preheater bypass line, and feedwater split flow bypass line.

10.4.7.2 - 3. Electrical Systems

Page 10.4-33: After "The split flow bypass valves," "feedwater split flow bypass valves" and "preheater bypass valves" insert "(Unit 2 only)."

10.4.7.3 - Safety Evaluation

Page 10.4-34: Insert the following paragraph- "The Unit 1 Delta 76 Feedring RSGs incorporate a feedring design in lieu of a preheater design and this feedring design feature prevents or mitigates the possibility of a steam generator water hammer event. (CP1-RSG-05-184)."

Page 10.4-34 & 35: Insert reference to Unit 2 on the paragraphs discussing waterhammer.

10.4.7.5 - Instrumentation Requirements

Page 10.4-37: Insert the following-"Based on section 10.4.7.3 Unit 1 does not require any instrumentation to mitigate feedline water hammer. The Unit 1 feedwater system is operated in a different manner after RSG installation. The FIV remains open during startup and a small flow through the valve and not the FIBV is used for purging purposes."

10.4.7.5 - 3. Feedwater Split Flow Bypass Valve (FSBV).

Page 10.4-39: Second sentence after "open" insert "at about 30 percent RTP due to system hydraulics" and delete "ever" from "whenever."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

10.4.9.1 - Design Bases

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

Page 10.4-45: Second paragraph first sentence insert "1106 psia (Unit 1) or" after "100 psia to" and insert "(Unit 2)" after "1107 psia."

Page 10.4-46: Insert the following at the end of Section 10.4.9.1:

Unit 1

"On Unit 1 the steam generators have been replaced with a model Delta 76 that employs a feedring design that eliminates the preheater. Auxiliary feedwater and feedwater entering the steam generator merge with the saturated liquid removed by the moisture separators. The combined feed then flows down the annulus formed by the steam generator shell and the tube bundle wrapper where it enters the tube bundle.

To ensure that steam does not flow back through either the main and auxiliary feedwater nozzles, the steam generator water level should be above the feedring at all operating levels where steam formation can occur. The operator is cautioned to maintain the normal SG level above the feedring at normal operating conditions. The normal operating Narrow Range level indication of 60% to 75% would result in sufficient water inventory to maintain the feedring and the AFW vertical perforated spray pipe submerged during the majority of expected transients like +/-10% load changes, turbine synchronizations, and plant loadings/unloadings. The supply piping to the auxiliary nozzle is designed with a loop seal immediately upstream of the SG nozzle to prevent steam backleakage.

In the event that steam does flow back past the loop seal and the check valves in the auxiliary feedwater line to the upper nozzle, the temperature of this line will increase causing temperature elements in the line to alarm in the control room. This allows the operator to take action to resolve the problem.

Unit 2

The Auxiliary Feedwater System is designed to preclude the effects of hydraulic instability due to water hammer by supplying water to the secondary side of the steam generator through a separate upper auxiliary feedwater nozzle. The Unit 2 preheater design steam generators permit the cold auxiliary feedwater to be heated as it comes down the side of the steam generator prior to reaching the feedwater preheater. For further discussion of the feedwater system upper nozzle arrangement and considerations see Subsection 10.4.7.

To ensure that steam does not flow back to the auxiliary feedwater nozzle, the steam generator water level should be above the auxiliary feedwater discharge pipe at all operating levels where steam formation can occur. Therefore, the operator is cautioned to maintain SG level above the auxiliary nozzle internal pipe extension when the temperature is above 212oF. This pipe is designed with a loop seal immediately upstream of the SG nozzle to prevent steam backleakage. Forward flushing is provided through the auxiliary feedwater nozzle virtually at all loads by AFW flow at startup

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued)

conditions, by preheater bypass flow during initial power ascension and by the split flow of the feedwater at RTPs above 30% full load.

In the event that steam does flow back past the loop seal and the check valves in the feedwater line to the upper nozzle, the temperature of this line will increase causing temperature elements in the line to alarm in the control room. This allows the operator to take action to resolve the problem.

10.4.9.2 - System Description

Page 10.4-47: Delete the paragraph beginning with "Downstream of the last isolation valve....." t

Page 10.4-48: Insert the following at the end of Section 10.4.9.2-

Unit 1

Downstream of the last isolation valve, each line from the motor-driven pumps joins with a corresponding line from the turbine driven pump to form a common line that connects with the feedwater line that connects to the auxiliary nozzle on the Unit 1 steam generator."

Unit 2

Downstream of the last isolation valve, each line from the motor-driven pumps joins with a corresponding line from the turbine driven pump to form a common line that connects with the feedwater preheater bypass line. The preheater bypass line connects to the Unit 2auxiliary nozzle on the steam generator."

10.4.9.3 - Safety Evaluation

Page 10.4-50: Insert "For Unit 2" at the start of the last paragraph

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Figure 10.4-12 - Auxiliary Feedwater Failure Mode Analysis Flow Diagram

FSAR Amendment 101, Supplement a

LDCR-SA-2006-38, EVAL-2003-002426-18 (JCH) (continued):

A dotted line was placed around the main feedwater piping to the SG to the preheater bypass piping and back to the main feed system identification block and text added that is "For Unit 2 Only."

Figure 10.4-22 - Main Feedwater Loop Seal Arrangement

Figure 10.4-22 was revised for Unit 1 and Unit 2. Figure 10.4-22A is for Unit 1 and is a new Figure. Figure 10-4-22B is the current figure with the title block revised to show "Unit 2" only and the a "B" added to Figure 10.4-22.

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

DESCRIPTION:

Figure II.E.1.1-1 - Auxiliary Feedwater System Simplified Flow Diagram

A dotted line was place around the main feedwater piping to the SG to the preheater bypass piping and back to the main feed water identification block and text added that is "For Unit 2 Only."

JUSTIFICATION:

Reflects system design modification made by the Steam Generator Replacement Project as defined in FDA-2005-000224-01, -02, -03, -04, -05, -06 and FDA-2003-002426-01. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and /or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS):

9.1.4.2.2 - Refueling Procedure

Item 2 - Phase II Reactor Disassembly

Insert the following sentence before the current first sentence: "The external air duct is removed from the CRDM Ventilation System's Air Handling Units and plenum (Unit 1 only)."

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Current first sentence after "shield" insert "(Unit 2 only)." Current second sentence after "Air ducts" insert "(Unit 2 only)."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

9.1.4.3.1 - Safe Handling

op paragraph next to last sentence insert "an integral steel missile shield (Unit 1 only) and a" after "employs" and insert "(Unit 2 only)" after current "shield.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.7N.1.3 - Critical Damping Values

Clarification added that the damping values given apply to both Units 1 and 2.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

4.5.1.1 - Materials Specifications

The Unit 1 RSG uses different stainless steel than Unit 2. Therefore, under "Pressure vessel" insert "Type F304LN for Unit 1," before "Type" and insert "for Unit 2" after "304."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

TMI II.F.2.3 - Reactor Coolant System Temperature

This revision indicates the differences between Unit 1 and Unit 2 in the location of the complete core exit thermocouples (CET) cable train separation. Therefore, the following changes are identified: In the next to last sentence on Page II.F-5 insert "CRDM Seismic Support Assembly's platform (Unit 1) and the missile" and insert "(Unit 2)" after "shield."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 17A-1 - List of Quality Assured Structures, Systems and Components

After "Control rod drive mechanism…" insert a new row as follows: "CRDM air cooling shroud assemblies (Unit 1 only), NNS, Mfrd Stds, - , II, Note E, and Note 10" respectively.

After "CRDM air cool baffle assemblies" insert "(Unit 2 only)."

After "Roll-away missile shield" insert "(Unit 2 only)."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.8.3.4.1 General Analysis of Internal Structure

Section revised to indicate that the STRUDL computer program was used for the evaluation of the Unit 1 steam generator compartments.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Table 3.8-1 Summary Table for Category 'I' Structures

Table was revised to include the latest stresses for the Unit 1 Reactor Building and steam generator upper lateral support beam. Revised loads due to the Replacement Steam Generator necessitated that these structures be re-analyzed.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.3.1.2 - Special Processes Used for Manufacturing and Fabrication

Divide Item 3 into a Unit 1 and Unit 2 section as follows: Insert paragraph "A: Unit 1" and the following: "The surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts." Before the existing sentence insert "B: Unit 2."

Divide Item 6 into a Unit 1 and Unit 2 section as follows: Insert paragraph "A: Unit 1" and the following: "The location of full penetration weld seams in the vessel bottom head are restricted to areas that permit accessibility during inservice inspection. There are no full penetration welds in the upper closure head." Before the existing sentence insert "B: Unit 2."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.3.3.1 - Design

First paragraph delete "The reactor vessel closure head" and insert the following the following for Unit 1 and Unit 2: Insert "A: Unit 1" and the following paragraph "The Unit 1 reactor vessel closure head contains head adapters. These head adapters are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these allow for welding to the control rod drive mechanisms. There are also two penetrations for the Reactor Vessel Level Measuring System (RVLMS) and four penetrations for the Core Exit Thermocouple Nozzle Assemblies (CETNA)." Insert "B. Unit 2" and "The Unit 2 reactor vessel closure head" before the existing "contains head adapters."

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.3.3.5 - Shipment and Installation

After the first paragraph insert Unit 1 and Unit 2 paragraphs as follows: Insert "A. Unit 1" and the following paragraph: "The Unit 1 replacement closure head is shipped with a protective shipping cover and skid. Two support plates within the shipping cover protect the control rod drive mechanism housings. All head openings are sealed to prevent the entrance of moisture and an adequate quantity of desiccant bags are placed inside the head. In addition, nitrogen purge is maintained to protect the control rod drive mechanism internals. The shipping cover also acts as a lifting frame for handling the replacement closure head." Inset "B. Unit 2" and then insert "Unit 2" before "closure head" of the existing paragraph.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.3.3.7 - Inservice Surveillance

The HAUP design does not include a knuckle transition piece. Therefore revise as follows: Second paragraph third sentence insert "For Unit 2," and delete "T" and insert "t" at the beginning of the sentence.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.3.3.8 - Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations (GL 97-01)

The Unit 1 RRVCH uses material Alloy 690. Therefore insert the following after the first paragraph making the remainder of the section applicable to Unit 2: "A: Unit 2 (Unit 1 RRVCH replaced Alloy 600 with Alloy 690)."

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.9N.4.1 - Descriptive Information of CRDS

ivide item 1 first paragraph into a paragraph for the Unit 1 RSG and Unit 2.

Insert "A. Unit 1" and the following paragraph" "For Unit 1, the pressure vessel is a one-piece housing which is connected to the reactor vessel head adapters by a full penetration weld. The lower portion of the pressure vessel contains the latch assembly. The upper portion of the pressure vessel provides space for the drive rod during its upward movement as the control rods are withdrawn from the core."

Item 2 second paragraph after "housing" in both sentences insert "/pressure vessel."

Second paragraph after item 4 insert "A. Unit 1" and the following paragraph "The Unit 1 control rod drive mechanism is welded with a full penetration weld to an adaptor on top of the reactor vessel and is coupled to the rod cluster control assembly directly below." Then insert "B. Unit 2" and insert "Unit 2" after "The" in the existing first sentence.

Fifth paragraph after item 4 after "housing" insert "/pressure vessel."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.9N.4.3.3 - Latch Assembly and Coil Stack Assembly

Unit 1 RSG and Unit 2 have different stainless steel materials.

Under "Latch Assembly - Thermal Clearances" first sentence insert "Type F304LN (Unit 1) or" before "Type 304" and insert "(Unit 2)" after "Type 304."

First, second and third paragraph under "Coil Stack Assembly - Thermal Clearances" after "latch housing" insert "/pressure vessel."

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.9N.4.1 - Descriptive Information of CRDS

Divide item 1 first paragraph into a paragraph for the Unit 1 RSG and Unit 2.

Insert "A. Unit 1" and the following paragraph" "For Unit 1, the pressure vessel is a one-piece housing which is connected to the reactor vessel head adapters by a full penetration weld. The lower portion of the pressure vessel contains the latch assembly. The upper portion of the pressure vessel provides space for the drive rod during its upward movement as the control rods are withdrawn from the core."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

6.2.2.3.3 - Recirculation Sump Design

Clarification of location of the Unit 1 RSG and respective piping accelerations. Page 6.2-38, third paragraph, after first sentence insert "As a result of the Steam Generator Replacement Project, the accelerations for the Unit 1 steam generators, the new main feedwater piping and the new auxiliary feedwater piping are location specific.."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Appendix 1A(B) -- Regulatory Guide 1.75

The discussion regarding compliance with Regulatory Position C.9 of Regulatory Guide 1.75 was modified to include the CRDM cable splices being installed in cable trays as part of FDA-2004-002710-02. In addition, a reference was added to Section 8.3, which addresses reduced separation requirements for instrumentation cables.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

8.3.1.4 - Independence of Redundant Systems

A paragraph was added to state that, in plant areas free from potential hazards such as missiles, external fires, and pipe whip and in non-hazard areas, the minimum separation the minimum separation between redundant instrumentation cables/trays is one inch horizontally and three inches vertically.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

3.7N.1.3 - Critical Damping Values

Clarification added that the damping values given apply to both Units 1 and 2.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 3.9N-20 - CRDM Head Adaptor Bending Moments

Actual versus allowable bending moments for the new Unit 1 CRDM Head Adaptors are provided in this Table

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Table 5.3-1 - Reactor Vessel Quality Assurance Program

Clarification added that for Unit 1, the closure head is a one-piece forging.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

7.7.1.9.1 - Thermocouples

Insert sentence after "......assemblies" "The Unit 1 thermocouples are provided with a grafoil seal and swage type seal from conduit to head." After "The" now third sentence insert "Unit 2."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 5.3-5 - Reactor Vessel Design Parameters

Under "Overall height of vessel….." insert the Unit 1 RSG value and identify the Unit 2 value as follows: Insert "43-6.4 (Unit 1)" and after "-10" insert "(Unit 2)."

Under "Closure head thickness (in.)" insert the Unit 1 RSG value and identify the Unit 2 value as follows" Insert "7 (Unit 1)" and after "-1/2" insert "(Unit 2)."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 5.3-15A - Unit 1 Reactor Vessel Non-Beltline Weld Metal Toughness Properties

The HAUP closure head design is not part of this table.

Delete the first four rows of this table for the HAUP. The closure head is a one piece design.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 5.3-2A - Unit 1 Reactor Vessel Fracture Toughness Properties

Delete the first three rows and insert the following row as row one : "Closure Head Forging, A508, C3, - 0.05, -, 0.87, -50, 10, -50, - and 184.0 respectively.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 9.4-2 - Design Conditions - Indoor

The "CRDM Shroud (Air Temperature)" at the inlet and the outlet remain the same for the HAUP/AHU and Unit 2, however Table 9.4-2 is revised to show that the HAUP/AHU inlet temperature is measured at the AHU inlet and not at the CRDM shroud as for Unit 2. Therefore, under "Reactor Containment Bldg. (RCB)" revise item 3 first column as follows: Insert "163 (CRDM Air Handling Unit inlet) (Unit 1)" before existing "163 (outlet)" and insert "(Unit 2)" after "163 (outlet)."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

5.2.3.4.4 - Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Clarification added that CRDM seal welds only apply to Unit 2.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Figure 3.9N-5 - Full Length Control Rod Drive Mechanism

The Unit 1 CRDM drive mechanism is different than the Unit 2 drive mechanism. A new figure has been developed for Unit 1 with the same title as the existing figure and the figure number has been changed to "Figure 3.9N-5A." For the existing figure delete "S 1 and" and insert "B" after the figure number in the title block for the Unit 2 CRDM.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 3.9N-6 - Full Length Control Rod Drive Mechanism Schematic

The Unit 1 CRDM drive mechanism schematic is different than the Unit 2 drive mechanism schematic. A new figure has been developed for Unit 1 with the same title as the existing figure and the figure number has been changed to "Figure 3.9N-6A." For the existing figure delete "S 1 and" and insert "B" after the figure number in the title block for the Unit 2 CRDM schematic.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Table 3.5-6 - Internally Generated Missiles (Inside Containment)

Due to the new CRDM design on Unit 1, the Unit 1 and Unit 2 missiles and type of protection are different. Therefore, revise as follows:

First row under "Identification of Missiles" insert "Unit 1," "Missile from CRDM drive shaft" and "Unit 2" before the existing text.

First row under "Missile Protection Provided" insert "Unit 1," "Integral steel missile shield above CRDM housing" and "Unit 2" before the existing text.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Figure 3.5-1 - Stationary Storage Scheme Roll-Away Missile Shield

This figure revised to show it is applicable to Unit 2 only. Insert "Unit 2 only" and "Unit 2" as shown on the markup.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 3.5-2 - Roll-Away Missile Shield

This figure revised to show it is applicable to Unit 2 only. Insert "Unit 2 only" and "Unit 2" as shown on the markup.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 1.2-8 - Building Cross-Section D-D

The Missile Shield and CRDM Cooling Unit have been deleted for Unit 1. Also, delete "Rev. No. 6" for drawing "A1-0529" and "Amendment 10, March 31, 1980."

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 1.2-13 - Primary Plant - Unit 1 Containment & Safeguard Buildings - Plans At EL. 852'-6" and 860'-0"

The indication of the position of the Stored Rolling Missile Shield has been deleted for Unit 1. For drawing A1-0503 delete "Rev. No. 8" and insert "Sheet -." Also delete all the "Roof Details" depicted on the left side of the figure for Unit 1. This information is now located on drawing "A1-503, Sheet A."

LDCR-SA-2006-39, EVAL-2003-002426-19 (JDS) (continued):

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 3.8-2 - Structural Plant Arrangement Section A-A

The Missile Shield above the reactor vessel has been deleted for Unit 1. Add note to Figure.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

Figure 3.8-15 - Containment Internal Structure

Notes added to Figure to indicate Missile Shield above reactor vessel is for Unit 2 only.

Reflects system design modification made by the Steam Generator Replacement Project (SGRP) as defined in FDA-2004-002710-02, FDA-2004-002710-03, FDA-2004-0002711-01 and FDA-2003-002426-02. These changes have been reviewed by the 50.59 process and were determined to not involve a change to an SSC that adversely affects an UFSAR described design function and/or that do not require prior NRC approval since the methodology had been previously approved by the NRC for the intended application.

LDCR-SA-2007-6, EVAL-2005-001957-21 (RJK):

Table 1.6-1, Section 4A and Section 15.6:

Update listing of approved topical reports by adding the following:

ERX-04-004-A; “Replacement Steam Generator Supplement To TXU Power's Large and Small Break Loss Of Coolant Accident Analysis Methodologies” Revision 0, March 2007.

ERX-04-005-A; “Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design” Revision 0, March 2007.

This is part of implementation of License Amendment 135.

FSAR Amendment 101, Supplement a

LDCR-SA-2003-21, EVAL-1999-000276-07 (JCH):

Table 3.9B-10

Changed valve type for 1DD-0020 and XDD-0103 from "diaphragm" to "ball."

Replacing the diaphragm valves with ball valves provides a more robust design that is unaffected by closure agsinst system pressure thus eliminating the maintenance issue and making the system more reliable.

LDCR-SA-2007-9, EVAL-2006-003910-01 (RJK):

Replace incorrect valve numbers HV-4872 and HV-4873 with valve numbers HV-4782 and HV-4783. Corrects typographic error which transposed the digits of the valve numbers.

FSAR Amendment 101, Supplement b

LDCR-SA-2006-1, EVAL-2005-003364-01 (RJK):

Section 12.3 - Barriers around the fuel transfer tube area will be removed from the plant. The primary reason to remove these barriers is to increase containment spray drainage flow to the containment sump. These non-essential metal barriers will be removed under FDA-2005-003364-07 (for Unit 2) and FDA-2005-003364-17 (for Unit 1) to support the response to GSI-191 concerns regarding containment sump operation. In order to enhance the flow from containment spray to the sumps during a postulated LOCA, these unnecessary wire mesh cages and doors will be removed.

Engineering analyses (via ME-CA-4000-0950) provided estimated dose rate projections for the RB 808' elev underneath the fuel transfer tube based on worse case source term assemblies and transient time. The results of this calculation define the dose rates at this location to be 187 mR/hr. EVAL-2002-003029-01 provides further assurance via typical field data from past fuel transfer tube activities that potential dose rates are less than that estimated in ME-CA-4000-0950 and less than the dose/dose rate limits prescribed in 10CFR20. As such, this is an acceptable change. Compliance with the requirements of TS 5.7 is provided administratively through the STAs. Therefore, removal of the barriers under the fuel transfer tube will not result in exposures to outage workers that exceed TS 5.7 limits and it is therefore an acceptable change.

LDCR-SA-2007-17, EVAL-2005-001957-20 (RJK):

Replace the reference to Figures 5.2-2 and 5.2-3 with "…the Pressure and Temperature Limits Report." Figures 5.2-2 and 5.2-3 are deleted from the FSAR.

The information contained in Figures 5.2-2 and 5.2-3 is present in the owner controlled Pressure and Temperature Limits Report, developed in accordance with Tech Spec 5.6.6. There is no reduction in control of the information. Maintaining the same information in two places is not necessary, so the information is being relocated to a more relevant document.

Delete Figure 5.2-2 from the FSAR.

The information contained in Figure 5.2-2 is present in the owner controlled Pressure and Temperature Limits Report, developed in accordance with Tech Spec 5.6.6. There is no reduction in control of the information. Maintaining the same information in two places is not necessary, so the information is being relocated to a more relevant document.

Delete Figure 5.2-3 from the FSAR.

The information contained in Figure 5.2-3 is present in the owner controlled Pressure and Temperature Limits Report, developed in accordance with Tech Spec 5.6.6. There is no reduction in control of the information. Maintaining the same information in two places is not necessary, so the information is being relocated to a more relevant document.

Add WCAP-14040 to the list of material incorporated in the FSAR by reference.

LDCR-SA-2007-17, EVAL-2005-001957-20 (RJK) (continued):

WCAP-14040 is the approved methodology as listed in Tech Spec 5.6.6 for development of the RCS H/U and C/D curves for the PTLR.

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE):

In section 1.2.2.3.5:

1. Delete all of 1.2.2.3.5 except of the last paragraph and insert the following before the last paragraph:

"Combustible gas control systems are not required to be Engineered Safety Feature Systems."

2. At the end of the last paragraph insert,

"This system purges the containment atmosphere through filters which reduce radioactive releases. See Section

6.2.5 for details."

Hydrogen recombiners are being removed consistent with the revised 10CFR50.44 and TSTF-447.

Delete the description of the hydrogen recombiners for regulatory guide 1.7 from 1A(N) and insert,

"Based on a revision to 10CFR40.44, Regulatory Guide 1.7 no longer applies to CPSES. See Section 6.2.5." Hydrogen recombiners are being removed consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete description of the hydrogen recombiners for regulatory guide 1.7 in 1A(B) and insert, "Based on a revision to 10CFR40.44, Regulatory Guide 1.7 no longer applies to CPSES. See Section 6.2.5." Hydrogen recombiners are being removed consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 3.1.2.7, delete discussion of the hydrogen removal system. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In sections 3.1.4.13 and 3.1.4.14, delete discussion of the hydrogen recombiners. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 6.2.5 to 6.2.5.3.1, delete discussion of the hydrogen recombiners. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE) (continued):

In section 6.2.5.3.3 to 6.2.5A.6, delete discussion of the hydrogen recombiners and hydrogen gas production. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Tables 6.2.5-1 and -2,

1. Delete Table 6.2.5-1 titled "Applicable Codes, Standards, and Regulatory Guides Used in the Design of the Electric Hydrogen Recombiner"

2.) Delete Table 6.2.5-2 titled "Electric Hydrogen Recombiner Parameters"

Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Table 6.2.5-4 for zinc corrosion rate. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete section 1 and 3 of Table 6.2.5-5 for FMEA for the hydrogen recombiners. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Table 6.2.5A-1 Sheets 1 to 3, Table 6.2.5A-2 Sheets 1 and 2, and Table 6.2.5A-3. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Table 6.2.5A-5 and Table 6.2.5A-6. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 7.1.1.1.2., delete discussion number 4 on combustable gas control. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

On Table 7.1-2.6 sheets 1 to 4, delete the column titled "Hydrogen Monitoring." Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 7.2.1.2.5, delete line item 3. for explosion from hydrogen buildup. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 7.3.1.2.5, delete line item 3. for explosion from hydrogen buildup. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 7.3.2.2, delete the third sentence which makes reference to GDC 41 for Hydrogen recombiners. Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE) (continued):

On Table 7.5-3, sheet 2, revise "Containment Hydrogen" Variable Function from "Key" to "Backup" and Type/Category from "B1" to "B3." On Table 7.5-4 revise "Containment Hydrogen" Variable Function from "Key to Backup" and Type/Category from "C1" to "C3". Removal consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Table 7.5-7A, sheet 3, revise category for containment hydrogen concentration from "B1, C1" to "B3, C3." Revision consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Identify deviation from RG 1.97 for hydrogen monitors.

On Table 7.5-7D, Sheet 7, at the bottom of the page starting from the first column add "Hydrogen Monitor," second column add "Category," third column add "Category 1," forth column add "Category 3," and the last column add "(24)."

Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Incorporate commitment to maintain hydrogen monitors capable of diagnosing beyond design bases accidents.

In Table 7.5-7E, Sheet 1 and 3,

1.) Remove the reference of "Containment Hydrogen Concentration" in numbers (3) on Sheet 1 and (19) on Sheet 3.

2.) Also on Sheet 3, iin number (19), second sentence, replace the word "are" with "is," and add "and License Amendment 117" at the end of number (24)

Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

On Table 8.1-1 sheet 1, delete reference to hydrogen recombiner and containment hydrogen purge system. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

For Table 8.3-1,

On Sheet 7, remove the reference to

1.) Electric H2 Recombiner Power Supply Panel (Post Accident) and

2.) Hydrogen Purge Exhaust Fan Filter Heater

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE) (continued):

On Sheet 8 of the same table above, remove the reference to

1.) Containment Hydrogen Purge Air Exhaust Fan.

On Sheet 9 of the same table above,

1.) Replace Specific Notes 12. and 13 with the words "Not used."

Include hydrogen recombiners to non class 1E equipment connected to safety related bus and footnote to assure remains disconnected during modes where the bus is required to be operable by technical specifications. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Table II.B.2-4 sheet 4, delete reference in number 15 to hydrogen recombiner. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete reference 15 to powering up hydrogen recombiner on Sheets 2, 4, 6, 7, 9, 10, and 12 of Table II.B.2-5. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete references 15A and 15B to powering up the hydrogen recombiner on Sheets 2, 5, 7, 9, 11, and 13 of Table II.B.2.-6. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section II.E.4.1 replace the CPSES Response with "Post accident combustible gas control vis external hydrogen purge or internal electric hydrogen recombiners is no longer required as described in Section 6.2.5. See Section 6.2.4 for containment penetration isolation." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section II.F.1.6, delete reference to hydrogen monitors as required and refer to section 6.2.5.2.3 and 7.5 for required commitment related to the monitors. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete the reference to "Electric Hydrogen Recombiner EQDP-SP-1" from Table 3.10N-1. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Chapter 6 Table of Contents, delete Sections, "6.2.5A.1 to 6.2.5A.6." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Chapter 6 List of Tables, replace the title of numbers, 6.2.5-1, 6.2.5-2, 6.2.5-4, 6.2.5A-1, 6.2.5A-2, 6.2.5A-3, 6.2.5A-5, and 6.2.5A-6 with the words "This Table has been deleted." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE) (continued):

In Chapter 6 List of Figures, replace figure titles of 6.2.5-2, 6.2.5A-1, 6.2.5A-2, 6.2.5A-3, and 6.2.5A-5 with the word "Deleted." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Chapter 6 List of Figures, replace the title of Numbers 6.2.5A-7, and 6.2.5A-9 with the words "Deleted." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Section 6.2.5.3.2.3 , delete the third paragraph which reads,

"After 30 days, the hydrogen generation rate from core radiolysis is 0.68 scfm. The generation rates from sump radiolysis and corrosion are 0.175 scfm, and 2.38 scfm, respectively."

Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

On page 6.2-80, delete the entire References. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Add the words "Abandoned in Place" at the top of Figure 6.2.5-1. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Figure 6.2.5-2. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Figures 6.2.5A-1, -02, and -03. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete the Figures 6.2.5A-5, -7, and -9. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In section 7.6.9,

1. replace the word "DBA" with "severe accident" and delete the word "System" in the first sentence of the first paragraph.

2.) delete the sentences in 7.6.9.1 and 7.6.9.2.

3.) delete the following words from the last paragraph, "System, which includes Hydrogen Monitoring, Hydrogen Recombiners and the Hydrogen Purge System."

Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

On Table 8.3-2, sheet 5, delete the Reference to "Electric H2 Recombiners Power Supply Panel. Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

LDCR-SA-2004-30, EVAL-2004-002063-03 (TJE) (continued):

On Table 8.3-2, sheet 12, replace Notes 12 and 13 with the words "Not used." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

From T 13.5-2 sheet 1, remove the reference to "Electric Hydrogen Recombiner System." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Section 15.6.5.4.5, remove the reference to "hydrogen purge system" and "electric hydrogen recombiners." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Table 17A-1, sheet 11, under number 9.a and in the 2nd and 3rd column, replace with "NA." Under the 5th column, replace the seismic category "I" with "II." Change the 6th column from "Note A" to "Note B." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

In Sections 6.2.5.3.2.3 and 6.2.5.3.2.4,

a. Relocate 6.2.5.3.2.3.b "Reactor Cavity" to 6.2.5.3.2.4.c,

b. Relocate 6.2.5.3.2.4.a "Pressurizer Subcompartment" to 6.2.5.3.2.3.b

c. Delete 6.2.5.3.2.3.c "Pressurizer Relief Tank Subcompartment" and insert the words into 6.2.5.3.2.4.a.

Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

Delete Figure and replace the title with "Deleted." Consistent with the revised 10CFR50.44 and TSTF 447. (SMF-EVAL-2004-002063-03)

LDCR-SA-2006-10, EVAL-2005-003364-03 (JDS):

Appendix 1A(B) RG 1.82 is revised to show conformance for new sump strainers. See EVAL-2005-003364-16 for the comparison of the old and new designs to the Regulatory Guide.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Table 3.9B-10 is revised to show the HV-4758/4759 are now butterfly valves.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

LDCR-SA-2006-10, EVAL-2005-003364-03 (JDS) (continued):

Section 6.2.2.2.1 is revised to describe the new sump strainer design and update the outdated description of emergency sump antivortex design found in FSAR Section 6.2.2.2.1 [SMF-2006-003988 issue].

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 6.2.2.3.3 is revised to describe the new sump strainer design and add the refueling cavity drain modifications.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 6.2.2.3.4 is revised to correct the NPSH description based on ME-CA-0232-5046 R0 which is the basis for Figure 6.2.2-2.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 6.2.2.5 is revised to describe the new RWST setpoint volumes. The containment water level instrument descritption is clarified to match the ref.d figures inlieu of the conrol board indicators.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Figure 6.2.2-3 is revised and Figure 6.2.2-3A added to show the new sump strainer design.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Figure 6.2.2-4 is revised to show the new refueling cavity drain strainers and debris screens.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Figure 6.2.2-5 is revised to show the removal of the pipe reducer in the refueling cavity drains.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

LDCR-SA-2006-10, EVAL-2005-003364-03 (JDS) (continued):

Section 6.3.2.8 is revised to reflect the new RWSTsetpoints and the new emergency strainer design. Typographical errors in valves tag numbers are corrected. Operator reset of SI is also clarified.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Table 6.3-7 is revised to reflect change in EOS-1.3 to start spray switchover after the Empty alarm based on ME(B)-389 R10.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Table 6.3-11 is revised to reflect change in the RWSToutflow analysis and ME(B)-389 R10 based on the above modifications.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 6.5.2.3.2 is revised to reflect the change in pH due to the RWST setpoint changes.

Section 6.5.2.2.1 is revised to clarify CAT isolation valves and vacuum breakers. [ACTN-SMF-2006-004073-01 and ACTN-SMF-2006-004097-01]

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Table 6.5-4 is revised to clarify CAT isolation valves and vacuum breakers. [ACTN-SMF-2006-004073-01 and ACTN-SMF-2006-004097-01]

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 7.6.5 is revised to show the instrument alarm uncertainty for new transmitters (changed from 2.5% to 2.3%).

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

Section 6.5.2.5.3 is revised to reflect some valves are motor and air operated.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

LDCR-SA-2006-10, EVAL-2005-003364-03 (JDS) (continued):

Table 17A-1 is revised to add the new strainers and debris screens.

Updates the FSAR for Modificartions to the plant in response to GSI-191 including License Amendment 129 [see SMF-2005-001869]

LDCR-SA-2006-31, EVAL-2006-001521-01 (JDS):

The sentence "The interlocks are redundant and can withstand a single failure." is being removed. This interlock is not redundant for system operation in normal mode, however, it satisfies the single failure criteria by being fail-safe rather than redundant. In addition, there is a manual override switch located on the fuel transfer system control panel. In case of mechanical failure of the toggle switch, this override feature can be utilized under administrative control. Therefore, redundancy in the upender clear interlocks is not required.

LDCR-SA-2007-19, EVAL-2001-002201-21 (JDS):

This type of insulation was not utilized and was not described in the insulation specification. In accordance with specification CPES-M-2012 Section 3.5.2 M -

Anti-sweat insulation used inside containment shall

1. be low to medium density fiberglass (e.g. < 8 lb/ft3)

2. not employ a vapor barrier, and

3. be stainless steel jacketed.

LDCR-SA-2007-23, EVAL-2007-002718-02 (GLM):

This proposed change is to add Design Basis Document, DBD-ME-006, to Chapter 1, Section 1.6, Table 1.6-1, DOCUMENTS INCORPORATED BY REFEFENCE.

DBD-ME-006, Attachment 1 captures the final response to NUREG-0612 and GL 81-07 as provided by TXX-3659. Documented in DBD-ME-006 are additional correspondence between the licensee and the NRC occurred after TXX-3659 was issued. This correspondence, including that from the NRC, form the basis for DBD-ME-006 content. By adding DBD-ME-006 to Table1.6-1, information that was provided by TXX-3659 to include subsequent correspondence with the NRC for heavy load controls (parameter values and assumptions, such as lift height, load weight and medium present under the load) would result in information describing the heavy load controls program that would normally be required in the FSAR (by the use of the Incorporated by Reference method). Any changes to the described heavy loads control program (changes to the DBD) have been and will be evaluated per the requirements of design change control process, as

LDCR-SA-2007-23, EVAL-2007-002718-02 (GLM) (continued):

well as 10CFR50.59 process. The original responses were provided by TXX-3659 [7.2.8]. DBD-ME-006 is maintained current under 10CFR50.59. By updating Table 1.6-1 to include DBD-ME-006 as incorporated by reference, and changing reference 17 to DBD-ME-006 (with historical listing of TXX-3659), information supporting the necessary details for FSAR content will be complete. This change has not affected on existing commitments. By implementation of this proposed change, the stated Regulatory Expections from RIS 2005-025 Supplement 1 will have been met.

Revise Section 9.1.4, Reference 17, to be DBD-ME-006 (with TXX-3659 being referenced for historical purposes).

DBD-ME-006, Attachment 1 captures the final response to NUREG-0612 and GL 81-07 as provided by TXX-3659. Documented in DBD-ME-006 are additional correspondence between the licensee and the NRC occurred after TXX-3659 was issued. This correspondence, including that from the NRC, form the basis for DBD-ME-006 content. By adding DBD-ME-006 to Table1.6-1, information that was provided by TXX-3659 to include subsequent correspondence with the NRC for heavy load controls (parameter values and assumptions, such as lift height, load weight and medium present under the load) would result in information describing the heavy load controls program that would normally be required in the FSAR (by the use of the Incorporated by Reference method). Any changes to the described heavy loads control program (changes to the DBD) have been and will be evaluated per the requirements of design change control process, as well as 10CFR50.59 process. The original responses were provided by TXX-3659 [7.2.8]. DBD-ME-006 is maintained current under 10CFR50.59. By updating Table 1.6-1 to include DBD-ME-006 as incorporated by reference, and changing reference 17 to DBD-ME-006 (with historical listing of TXX-3659), information supporting the necessary details for FSAR content will be complete. This change has not affected on existing commitments. By implementation of this proposed change, the stated Regulatory Expections from RIS 2005-025 Supplement 1 will have been met.

LDCR-SA-2007-22, EVAL-2004-002882-07 (JDS):

RG 1.54 description is revised to describe the revised program from Non-safety to Service Level I and the applicable standards. This is an administrative update of the FSAR to reflect an upgrade to the Protective Coatings Program for inside containment. Coatings do not have safety function per se and are not Basic Components as defined in 10CFR50.2. They are not subject to 10CFR21. In addition, they are not "Safety Related" Structures, Systems, or Components as defined in 10CFR50.2. However, coatings are subject to applicable portions of 10CFR50, Appendix B via GDC-1 because their failure has the potential to be detrimental to Safety Related Structures, Systems, and Components. This is most similar to non-seismic items (seismic category II equipment). The CPNPP quality assurance program for such items is covered by Appendix D of the QA Manual. See DBD-ME-028 for additional information on classification of SSCs. [Also note that this change is consistent with ONE 97-1016, Applicability of Appendix B to protective coatings applied to the interior surfaces of safety related components]

LDCR-SA-2007-22, EVAL-2004-002882-07 (JDS) (continued):

Updated description of the Protective Coatings Program for containment. This is an administrative update of the FSAR to reflect an upgrade to the Protective Coatings Program for inside containment.

Updated to remove obsolete information. This is an administrative update of the FSAR to reflect an upgrade to the Protective Coatings Program for inside containment.

Update and clarify description of the quality assurance program (under QA Manual Appendix D). This is an administrative update of the FSAR to reflect an upgrade to the Protective Coatings Program for inside containment. Coatings do not have safety function per se and are not Basic Components as defined in 10CFR50.2. They are not subject to 10CFR21. In addition, they are not "Safety Related" Structures, Systems, or Components as defined in 10CFR50.2. However, coatings are subject to applicable portions of 10CFR50, Appendix B via GDC-1 because their failure has the potential to be detrimental to Safety Related Structures, Systems, and Components. This is most similar to non-seismic items (seismic category II equipment). The CPNPP quality assurance program for such items is covered by Appendix D of the QA Manual. See DBD-ME-028 for additional information on classification of SSCs. [Also note that this change is consistent with ONE 97-1016, Applicability of Appendix B to protective coatings applied to the interior surfaces of safety related components]

LDCR-SA-2006-24, EVAL-2003-000188-03 (JCH):

Table 17A-1

Added notes to the table for manual and extension passive and active valves to provide for the installation of Safety Related / Seismic Category I replacement remote valve stem extensions and gear driven valve operators for the Atmospheric Relief Valves (ARV) upstream isolation block valves, and Seismic Category I supports.

The table lists valve operator types. Clarifying UFSAR Table 17A-1 to describe the requirements for Active valve manual and extension stem operator quality attributes and Passive valve manual and extension stem operator quality attributes is acceptable based on the different quality requirement attributes for active and passive components. This clarification also maintains consistency between the table requirements for safety related valves and the referenced Note 79. This clarification is supported by the requirements of DBD-ME-028, Rev. 13, Attachment 19, Table 2.

Table 3.9B-10

Provided editorial clarification of valve/operator type and method of operation descriptions by specifying handwheels or extension stems as appropriate

FSAR Amendment 101, Supplement b

LDCR-SA-2006-34, EVAL-2003-002117-03 (JCH):

Section 10.4.3.1-Design Basis-2. Feedwater Pump Turbine Seals

Delete "The feedwater pump turbine is self-sealing at loads above approximately half of the rated load on the turbine."

Section 10.4.3.2-System Description-1. Normal Operation Conditions-b. Feedwater Pump Turbine Steam Seals

Delete "during startup of the turbine and during low-power operation."

Justification: The FWPT technical manual shows that the originally supplied design anticipated seal steam would be supplied to the turbine during both start-up and full load operation; therefore, self sealing is not a design attribute of the FWPT's at CPSES. Operations has never seen a self-sealing indication for the feedwater pump turbine in either steam seal supply or dump valve positions. This does not affect the operational capability of the seal system and is congruent with the expectations outlined by the vendor.

LDCR-SA-2007-16, EVAL-2006-002471-02 (RJK):

The values presently listed in Section 5.3.2.2.1 are historic to pre-operational calculations performed IAW Reg Guide 1.99 revision 1 for pre-licensing certification of the reactor vessels. These values were recalculated just prior to Unit 1 and 2 startup using revision 2 of Reg Guide 1.99, but were never updated in the FSAR. These updated values are currently listed in the CPSES PTLR (Table 2-1) and were previously reported to the USNRC via letters TXX-95220 and TXX-97275 in response to Generic Letter 92-01.

LDCR-SA-2007-5, EVAL-2006-003933-05 (JDS):

The Unit 1 Cycle 13 core configuration consists of 88 fresh Westinghouse fuel assemblies (Region 15), 89 once-burned Westinghouse assemblies (88-Region 14 & 1-Region 13), and 16 twice-burned Westinghouse assemblies (Region 13). The single once-burned Region 13 fuel assembly is reloaded from the spent fuel pool. The Region 14/15 Westinghouse fuel assemblies are of the Vantage-Plus with Intermediate Flow Mixing (IFM) grids) and include a thin zirconium oxide coating applied to the bottom seven inches of each fuel rod to provide additional debris fretting protection in this area. Oxide forms naturally on the fuel cladding during normal operation so this coating has no impact on fuel design function. The Region 13 Westinghouse fuel assemblies are also of the Vantage-Plus design, but do not include IFMs or the applied oxide coating but are dimensionally equivalent to the Vantage-Plus design. All fuel assemblies are equipped with the “Small Hole” debris filtering bottom nozzle, an alternate protective grid (P-grid) located just above the bottom nozzle to trap any debris making it through the bottom nozzle, and long solid end plugs. The pressure drop through the Vantage-Plus (without IFMs) assemblies is slightly lower than that through the co-resident fresh Vantage-Plus / IFM assemblies while the heat transfer characteristics are improved by the IFMs. The effects of the mixed core on the thermal hydraulic response are considered.

LDCR-SA-2007-5, EVAL-2006-003933-05 (JDS) (continued):

All the Westinghouse assemblies use ZIRLO (as opposed to Zircaloy-4) for the fuel rod cladding, guide tubes and assembly mid-span grids. In addition, integrated fuel burnable absorbers (IFBAs) and discrete wet annular burnable absorbers (WABAs) are used to shape the power distribution and achieve a desirable moderator temperature coefficient.

The thermal-hydraulic and neutronic responses of the Unit 1 Cycle 13 reload core configuration potentially affect the design functions of many plant components and systems. The acceptability of the Cycle 13 reload core configuration is based on meeting the event acceptance criteria in FSAR Sections 3.6B, 4.2, 4.3, 4.4, 6.2, 6.3, and all of FSAR Chapter 15.

The neutronic design criteria of the Cycle 13 reload core configuration continue to be met. The thermal-hydraulic design criteria of the Cycle 13 reload core configuration continue to be met.

The adequacy of the systems required to mitigate or prevent accidents, shutdown the plant and maintain the plant in a safe shutdown condition is demonstrated in the analyses presented in FSAR Chapter 15 and summarized in the Cycle 13 update of FSAR Appendix 4A. The relevant event acceptance criteria include compliance with the DNBR and peak RCS and steamline pressure acceptance criteria for ANS Condition II transients and acceptable radiological consequences for ANS Condition III and IV accidents. Relevant event acceptance criteria continue to be met with the current cycle-specific reactor physics parameters.

LDCR-SA-2007-21, EVAL-2006-004074-01 (GLM):

Remove the paragraph in FSAR Section 9.2.1.2.1 which states that the Circulating Water Bleed line is used as a continuous makeup for SSI and add a note to Section 9.2.1.2.1 which states that additional details can be found in Section 9.2.5.2.

As stated in FSAR Section 9.2.5.1 the function of the ultimate heat sink is to dissipate heat rejected from the station service water system during post accident shutdown and normal cooldown conditions. The Safe Shutdown Impoundment (SSI) is a part of the Ultimate Heat Sink. The level of water in the SSI is maintained through an equalization canal between the Squaw Creek Reservoir (SCR) and the SSI as described in the FSAR in Sections 9.2.1.2.1 and 9.2.5.2. The equalization canal between the SSI and Squaw Creek Reservoir is current the normal source for makeup to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

A bleed line is provided from the Circulating Water System, at the intake structure, to the Service Water Intake Structure pump suction pit. The FSAR indicates that when operated the circulating water bleed-off ensures continuous makeup to the SSI and provides circulation to prevent excessive dissolved solids concentration in the SSI. Based on the operations procedures (SOP-310A and SOP-310B) above this is not

LDCR-SA-2007-21, EVAL-2006-004074-01 (GLM) (continued):

currently the way that the system is operated. The procedure indicates that the bleed line is normally closed. This is supported by the valve line-up attachment to the procedures which indicates that valve 1CW-0019 is normally closed.

IOR No. 89-64 was written to address environmental regulations which require that the discharge from the Safe Shutdown Impoundment to the Squaw Creek Reservoir be monitored. This includes the flow from the Circulating Water Bleed line. The IOR states that valve 1CW-0019 is intended to remain closed during plant operation until the plasitc is removed from the bleed line. Based on the changes made by IOR No. 89-64 and CPSES Procedure Change Form for SOP-310A, Rev. 5, the method of operation was revised in 1990. The IOR states that the flow Circulating Water System to the Safe Shutdown Impoundment via the bleed line be closed using valve 1CW-0019. In lieu of using the Circulating Water bleed line as the normal makeup, the equalization canal between the SSI and Squaw Creek Reservoir is used as the normal source for make-up to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

Delete the last paragraph in FSAR Section 9.2.5.2 and add a new paragraph describing the Circulating Water Bleed line and its current operation.

As stated in FSAR Section 9.2.5.1 the function of the ultimate heat sink is to dissipate heat rejected from the station service water system during post accident shutdown and normal cooldown conditions. The Safe Shutdown Impoundment (SSI) is a part of the Ultimate Heat Sink. The level of water in the SSI is maintained through an equalization canal between the Squaw Creek Reservoir (SCR) and the SSI as described in the FSAR in Sections 9.2.1.2.1 and 9.2.5.2. The equalization canal between the SSI and Squaw Creek Reservoir is current the normal source for makeup to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

A bleed line is provided from the Circulating Water System, at the intake structure, to the Service Water Intake Structure pump suction pit. The FSAR indicates that when operated the circulating water bleed-off ensures continuous makeup to the SSI and provides circulation to prevent excessive dissolved solids concentration in the SSI. Based on the operations procedures (SOP-310A and SOP-310B) above this is not currently the way that the system is operated. The procedure indicates that the bleed line is normally closed. This is supported by the valve line-up attachment to the procedures which indicates that valve 1CW-0019 is normally closed.

IOR No. 89-64 was written to address environmental regulations which require that the discharge from the Safe Shutdown Impoundment to the Squaw Creek Reservoir be monitored. This includes the flow from the Circulating Water Bleed line. The IOR states that valve 1CW-0019 is intended to remain closed during plant operation until the plasitc is removed from the bleed line. Based on the changes made by IOR No. 89-64 and CPSES Procedure Change Form for SOP-310A, Rev. 5, the method of operation was revised in 1990. The IOR states that the flow Circulating Water System to the Safe Shutdown Impoundment via the bleed line be closed using valve 1CW-0019. In lieu of

LDCR-SA-2007-21, EVAL-2006-004074-01 (GLM) (continued):

using the Circulating Water bleed line as the normal makeup, the equalization canal between the SSI and Squaw Creek Reservoir is used as the normal source for make-up to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

In Section 9.2.5.3, under "2. Hydraulic Performance," delete the word "creates" and replace with "can be used if required."

As stated in FSAR Section 9.2.5.1 the function of the ultimate heat sink is to dissipate heat rejected from the station service water system during post accident shutdown and normal cooldown conditions. The Safe Shutdown Impoundment (SSI) is a part of the Ultimate Heat Sink. The level of water in the SSI is maintained through an equalization canal between the Squaw Creek Reservoir (SCR) and the SSI as described in the FSAR in Sections 9.2.1.2.1 and 9.2.5.2. The equalization canal between the SSI and Squaw Creek Reservoir is current the normal source for makeup to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

A bleed line is provided from the Circulating Water System, at the intake structure, to the Service Water Intake Structure pump suction pit. The FSAR indicates that when operated the circulating water bleed-off ensures continuous makeup to the SSI and provides circulation to prevent excessive dissolved solids concentration in the SSI. Based on the operations procedures (SOP-310A and SOP-310B) above this is not currently the way that the system is operated. The procedure indicates that the bleed line is normally closed. This is supported by the valve line-up attachment to the procedures which indicates that valve 1CW-0019 is normally closed.

IOR No. 89-64 was written to address environmental regulations which require that the discharge from the Safe Shutdown Impoundment to the Squaw Creek Reservoir be monitored. This includes the flow from the Circulating Water Bleed line. The IOR states that valve 1CW-0019 is intended to remain closed during plant operation until the plasitc is removed from the bleed line. Based on the changes made by IOR No. 89-64 and CPSES Procedure Change Form for SOP-310A, Rev. 5, the method of operation was revised in 1990. The IOR states that the flow Circulating Water System to the Safe Shutdown Impoundment via the bleed line be closed using valve 1CW-0019. In lieu of using the Circulating Water bleed line as the normal makeup, the equalization canal between the SSI and Squaw Creek Reservoir is used as the normal source for make-up to the SSI as indicated by the Operations procedures SOP-310A and SOP-310B.

FSAR Amendment 101, Supplement b

LDCR-SA-2007-11, EVAL-2005-001957-23 (JDS):

Update list of Tables to match revised Tables.

Update list of Figures to Match revised or deleted Figures.

Sections 6.2.1.1.3.1.1 and 6.2.1.1.3.1.2 were modified due to the methodology change and Steam Generator replacement which resulted in differences in the analysis results for the two units.

Section 6.2.1.1.3.2 was modified to reflect the changes in the Figures and the RSGs.

Section 6.2.1.1.3.7 is modified to reflect the changes in the Tables and the RSGs.

Section 6.2.1.2.1 is modified to reflect the changes in the methodology and RSGs. Section 6.2.1.2.2 is modified to reflect the changes in the Figures.Section 6.2.1.2.3 is modified to reflect the changes in the methodology and the addition of the RSGs for Unit 1. Also, many of the Figures and Tables described in this Section were determined to be no longer required, and therefore removed.

Section 6.2.1.3 was re-written due to the methodology update.

Sections 6.2.1.4.1 and 6.2.1.4.2 is modified to reflect the changes or deletions in the Tables

Section 6.2.1.4.9 is deleted due to the deletion of the Tables described in this Section.

Section 6.2.1.5.6 is modified to reflect the changes in the Tables

Section 6.2.2.3.2 is modified to reflect the changes in the Tables

6.2.1-2A was replaced with 6.2.1-2A.1 and 6.2.1-2A.2 due to differences between Units 1 and 2 as a result of Unit 1 Steam Generator Replacement.

6.2.1-2B was replaced with 6.2.1-2B.1 and 6.2.1-2B.2 due to differences between Units 1 and 2 as a result of Unit 1 Steam Generator Replacement.

Tables 6.2.1-3 and -3A were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

LDCR-SA-2007-11, EVAL-2005-001957-23 (JDS) (continued):

Tables 6.2.1-4, -4A, and -4B were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-15 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

6.2.1-9 was replaced with 6.2.1-9A and 6.2.1-9B due to differences between Units 1 and 2 as a result of Unit 1 Steam Generator Replacement.

6.2.1-10 was replaced with 6.2.1-10A and 6.2.1-10B due to differences between Units 1 and 2 as a result of Unit 1 Steam Generator Replacement.

Table 6.2.1-21 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-26 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-29 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

LDCR-SA-2007-11, EVAL-2005-001957-23 (JDS) (continued):

Table 6.2.1-30 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Tables 6.2.1-37 through -49 were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-50 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-50A was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-51 through -56 were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-58 through -60 were deleted as the information contained in ithem is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

FSAR Amendment 101, Supplement b

LDCR-SA-2007-11, EVAL-2005-001957-23 (JDS) (continued):

Table 6.2.1-64 through -69 were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-77 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-91 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-93 was deleted as the information contained in it is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Table 6.2.1-95 through -104 were deleted as the information contained in them is not required. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-1 through -3 were deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-9 through -16 were modified due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

LDCR-SA-2007-11, EVAL-2005-001957-23 (JDS) (continued):

Figure 6.2.1-20 was deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-30 through -45 were deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-49 through -52 were deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-54 through -56 were deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

Figures 6.2.1-58 through -70 were deleted due to the revised analysis. The Mass and Energy Release Data for both LOCA and SLB for Unit 2 is unchanged. However, for Unit 1, the mass and energy release data for both LOCA and SLB have been modified. For LOCA, the change in the M&E release data is due to both the RSGs and a methodology change. For SLB, the change in the M&E release data is due solely to the RSGs.

FSAR Amendment 102

LDCR-SA-2007-18, EVAL-2003-000167-04 (RJK):

Neutron dosimetry sensors used to monitor neutron exposure for continued surveillance of reactor vessel integrity were installed in CPNPP unit 1 reactor cavity annulus per FDA 2003-000167-01 and the FSAR was updated per LDCR SA-2003-059. This LDCR was required to document that the neutron dosimetry sensors are also now installed in the unit 2 reactor cavity annulus and FSAR section 5.3.1.6.2 now applies to both units.

LDCR-SA-2007-20, EVAL-2006-000629-04 (TJEW):

In 9.1.3.1.1.1, the first sentence of the second paragraph reads, "The spent fuel pool bulk water temperatures are maintained at less than 150EF for normal operation based on decay heat generation from a normal full core offload at 7days after shutdown, plus decay heat from the opposite unit's last refueling discharge plus decay heat from fuel assemblies from a maximum number of previous refuelings in both pools."

1.) In the 150EF, replace the "E" with a degree symbol, and

2.) Change "7 days" to "125 hours".

In 9.1.3.1.1.1, the third and fourth sentence in the fourth paragraph reads, "The start of the second outage is conservatively assumed to begin 45 days after the first. A normal full core offload is conservatively assumed to start at 100 hours after the reactor is subcritical and complete at 168 hours."

1.) Change "45 days" to "4 months",

2.) Change "100" hours to "75" hours, and

3.) Change "168" hours to "125" hours.

4.) In the last paragraph of 9.1.3.1.1.1, for the term "140EF" replace the "E" with a degree sign.

In 9.1.3.1.1.3, the second paragraph reads, "The spent fuel pool water temperatures are maintained at less than 212°F for two loop operation based on an emergency core offload 150 hours after shutdown, plus the most recent refueling discharge 36 days after shutdown, plus the opposite unit's previous refueling discharge 66 days after shutdown, plus decay heat from a maximum number of previous refuelings in both pools."

1.) Change "150" hours to "125" hours,

2.) Change "36 days after" to "31 days prior to", and

3.) Change "66 days after" to "5 months prior to"

1.) On Table 9.1-1, under Parameters,

LDCR-SA-2007-20, EVAL-2006-000629-04 (TJEW) (continued):

a.) delete the row "Numbr of fuel assemblies stored" and

b.)change the exponent of Decay Heat Producted from "10 to the 5" to 10 to the 6"

2.) On the same row of Decay heat produced, make the following changes:

Under MAX. DESIGN

a.) MAX POOL from 50.7 to 52.0

b.) MIN POOL from 5.55 to 9.09

Under MAX. SUMMER DESIGN

a.) MAX POOL from 21.3 to 23.28

b.) MIN POOL from 5.55 to 6.08

Under ABNORMAL MAX. DESIGN

a.) MAX POOL from 56.5 to 62.92

b.) MIN POOL from 6.0 to 6.6

3.) For the parameter Time to boil, change MAX SUMMER DESIGN, MAX POOL from >7 to >3

4.) Under the "NOTES" section,

a.) Replace the words in the note with “Deleted”, and

b.1) in note 3, replace the words "assumes most resent spent fuel discharges are" with the word “loads”, and

b.2) delete "to" after the words "MAX Pool"

On Table 12.2-24, "DESIGN PARAMETERS FOR POST-ACCIDENT DOSE EVALUATIONS"

1.) Under Gap activity-release

a.) Replace "Noble gases other than Kr-85 0.1" with "I-131 0.08"

b.) Change the value of Kr-85 from "0.3" to "0.10"

LDCR-SA-2007-20, EVAL-2006-000629-04 (TJEW) (continued):

c.) Replace "Iodines other than I-127 and I-129 0.1" with "Other Iodines and Noble Gases 0.05"

2.) Change the Decay interval between reactor shutdown and commencement of refueling operations (hours) from "100" to "75"

In 15.7.4.2.1, change "100 hr" to "75 hours"

In 15.7.4.3.1.1, change "100 hr" to "75 hours"

1.) In 15.7.4.3.1.3, change "100" hours to "75" hours

2.) In the last paragraph of 15.7.4.3.1, change "0.087" rem to "0.090" rem

Revise Table 15.7-6, NOBLE GAS AND IODINE ACTIVITIES RELEASED FROM DAMAGED FUEL RODS AS A RESULT OF A FUEL HANDLING ACCIDENT as follows:

1.) Revise Kr-85 from "8.62" to "9.66"

2.) Add "Kr-85m" to be "9.62E-02" curries released

3.) Add "Kr-88" to be "3.05E-04" curries released

4.)Revise Xe-131m from "3.01" to "4.57"

5.) Revise Xe-133m from "5.56" to "1.52"

6.) Revise Xe-133 from "6.93" to "6.88"

7.) Revise Xe-135m from "3.61" to "5.04E+00"

8.) Revise Xe-135 from "1.20" to "7.05"

9.) No revision to I-131

10.) Revise I-132 from "2.87" from to 3.25

11.) Revise I-133 from "3.47" to "7.44"

12.) Revise I-135 from "2.36" to "3.07E+01"

13.) Revise a.3 from "100" hours to "75" hours

On Table 15.7-8, PARAMETERS FOR POSTULATED FUEL HANDLING ACCIDENT ANALYSIS, change

LDCR-SA-2007-20, EVAL-2006-000629-04 (TJEW) (continued):

1.) In 1.a, Power Level (MWt) from "3565" to "3694" and

2.) In 4.c, Doses, change "0.087" rem to "0.090" rem

LDCR-SA-2006-41, EVAL-2006-003110-02 (CBC):

In paragraph 6.2.6.4, item 1, revise the first by inserting an "s" at the end of "exception" and deleting the numeral "1" at the end of "5.5.16.a.1". The new sentence will read as follows:

"Containment integrated leakage-rate test (Type A) are performed at least once per 48 months with possible extension to 10 yr. intervals based upon acceptable performance history as determined in accordance with NEI 94-01 and Reg. Guide 1.163, as modified by the exceptions in Technical Specification 5.5.16.a."

FSAR Section 6.2.6.4 is updated by eliminating the concrete containment inspections required by Appendix J, Option B and taking credit for the concrete containment inspections performed under the IWL and IWE of the ASME program (which CPSES is already committed to). The FSAR is updated to be consistent with changes to TS 5.5.16 approved by the NRC in License Amendment 141.

LDCR-SA-2006-23, EVAL-2005-004439-02 (JCH):

Page 9.5-71: add "wireless intercom system" to the types of communication systems listed

Justification: The Refuleing and Emergency sound Powered Loops have been replaced by a wireless intercom with radio backup and emergency communications is supplied by the Inplant Radio System with Gaitronics backup.

Delete references to refueling and emergency sound powered loops

Justification: The Refuleing and Emergency sound Powered Loops have been replaced by a wireless intercom with radio backup and emergency communications is supplied by the Inplant Radio System with Gaitronics backup.

Added new section 9.5.2.2.8 which discusses the wireless intercom system and renumbered subsequent sections.

Justification: The Refuleing and Emergency sound Powered Loops have been replaced by a wireless intercom with radio backup and emergency communications is supplied by the Inplant Radio System with Gaitronics backup.

Added paragraph that power to the wireless intercom system is provided by plant support power and each headset has an individual battery supply.

LDCR-SA-2006-23, EVAL-2005-004439-02 (JCH) (continued):

Justification: The Refuleing and Emergency sound Powered Loops have been replaced by a wireless intercom with radio backup and emergency communications is supplied by the Inplant Radio System with Gaitronics backup.

Deleted reference to testing of refueling and emergency sound powered loops

Justification: The Refuleing and Emergency sound Powered Loops have been replaced by a wireless intercom with radio backup and emergency communications is supplied by the Inplant Radio System with Gaitronics backup.

LDCR-SA-2005-29, EVAL-2005-003364-07 (JDS):

Suction pressure accident monitoring is added for the RHR and Containment Spray pumps. Table 7.5-7B is updated. Flow Diagrams 5.4-6 and 6.2.2-1 are updated as detailed in FDA-2005-003364-08 and -18. These modifications will be implemented at power in 2007.

Suction pressure accident monitoring is added for the RHR and Containment Spray pumps.These components have been identified in Attachment 1 of TXX-05162 as requiring monitoring for system performance. The are utilized as a key variable for emergency sump performance. These instruments are not required by Table 2 of RG 1.97 Rev. 2. CPSES utilization is an alternate to narrow range sump level, Type B Category 2. It is classifed as a Type D Category 2 variable because it provides information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

LDCR-SA-2008-4, EVAL-2007-001435-01 (RJK):

Changes system description to allow use of both CVCS mixed bed demineralizers during RCS cleanup during outages. FDA-2007-001435-01 provides the basis for allowing increased flow during RCS cleanup based upon less than minimal impact on SSCs and radiological consequences due to higher flow rates.

The FSAR is also administratively updated to clarify the maximum letdown flow. DCN 9970 provides the basis for the normal CVCS flow for a nominal 120 gpm design is a maxiumum of 140 gpm based on accuracies of the orifice construction. This flow rate corresponds to a critical input to the radiological consequences in Section 15.6.2.

The FSAR is updated to clarify the maximum letdown flow. DCN 9970 provides the basis for the normal CVCS flow for a nominal 120 gpm design is a maxiumum of 140 gpm based on accuracies of the orifice construction. This flow rate corresponds to a critical input to the radiological consequences in Section 15.6.2.

LDCR-SA-2008-4, EVAL-2007-001435-01 (RJK) (continued):

Section 15.6.2 is administratively corrected to note that the pressure switches are not safety related and may NOT be credited as "the" method of detection.

LDCR-SA-2006-3, EVAL-2001-001214-01 (JCH):

Section 10.2.2.6 Generator

Sectin 10.2.2.6.2 Fire Prevention and Protection

Page 10.2-6: Change the referenced inert gas from "carbon dioxide" to "argon".

Justification: Change of the inert gas used when purging the generator was changed from carbon dioxide to argon due equipment obsolescence, personnel safety and improved outage purging time.

LDCR-SA-2007-2, EVAL-2005-002580-04 (TJEW):

Revise turbine HP & LP stop and control valves test frequency from 12 weeks to 26 weeks. This recommendation was based on a quantitative evaluation on the probability of failure of the overspeed trip and protection system as a function of the turbine stop and control valve test interval. The basic principles of the methodology are the same as those used by Siemens in previous studies which included the CPSES original designs, Grand Gulf, Connecticut Yankee, and Limerick and which have been previously reviewed and accepted by the NRC.

For Table 1.6-1, sheet 2 of 3, replace the 9th Reference which reads:

"'Missile Probability Analysis Methodology for TXU Generation Company, LP, Comanche Peak Units 1 & 2 with Siemens Retrofit Turbines,' CT-27331 (CONFIDENTIAL), Revision No. 3, December 15, 2003"

with the following words,

"TP-03143, "Missile Analysis Methodology for GE Nuclear Steam Turbine Rotors by the SWPC," July 31, 2003."

In Section 3.5.1.3.2,

1.) In the 1st sentence in the 4th paragraph, replace the word "probabiliy" with "probability"

2.) In the 5th paragraph change the Section number from "3.5.1.3.3.1" to "3.5.1.3.3" and delete the words "Unit 1"

In Section 3.5.1.3.3, add "[17]" to the end of the first paragraph.

LDCR-SA-2007-2, EVAL-2005-002580-04 (TJEW) (continued):

1.) In section 3.5.1.3.3 on page 3.5-9, replace the words,

"P1 = 3.42E-5 per year for 100,000 hour inspection interval [15]"

with the words,

"P1 = 1.56E-4 at 100,000 hours (inspection interval) for a 26 weeks turbine valve test frequency [15]"

2.) In the last pargrapgh second sentence after "P1" add the following works, "at 100,000 hours inspection interval" and after "the NRC limit of" add the following words, "11.42E-4 when adjusted to 100,000 hours inspection interval verses".

3.) In section 3.5.1.3.4, number 2, change "12" to "26"

4.) In section 3.5.1.3.4 in the last paragraph, change "5x10^-6" to "1.37x10^-5" and

replace References " [5,11,13,][14]" with the following words, "and 26 week turbine valve test interval [15]"

5.) Delete the last sentence of the last paragraph that reads, "CPSES turnine missile analysis is based on a conservative high turbine missile generation probability of 4 x 10^-5 per turbine year."

6.) In section 3.5.1.3.5, replace the following sentence that reads,

"Periodic testing of the stop and control valves is necessary to insure reliability and continuity of service; therefore, valve testing for all nuclear units is recommended with the aid of an automatic turbine tester (ATT) every 12 weeks."

with the following sentence,

"Periodic testing of the stop and control valves is necessary to insure reliability and continuity of service; therefore, valve testing for all nuclear units is recommended with a manual test or with the aid of an automatic turbine tester (ATT) every 26 weeks."

At the end of the first paragraph and at the end of the 17 th of 10.2.3.5, replace Reference [10] with [2].

In the Reference section of 3.5,

1.) replace References 5 with the word "Deleted"

2.) replace References 11, 13 and 14 with the word "Deleted"

LDCR-SA-2007-2, EVAL-2005-002580-04 (TJEW) (continued):

3.) change the revision number of Reference 15 from "4" to "5," change the date from "December 15, 2003" to "January 18, 2007," and delete the VL number.

1.) For Reference 3.5, delete the VL number from Reference 16.

2.) For Reference 3.5, add Reference 17 that will read

"TP-03143, "Missile Analysis Methodology for GE Nuclear Steam Turbine Rotors by the SWPC," July 31, 2003."

In section 10.2.2.1, first paragraph, change "44" to "46."

In section 10.2.2.7.6, number 2, change "12" to "26."

In the 4th paragraph of 10.2.3.6, change "12" to "26."

1.) In the Reference section of 10.2, replace the VL numbers in Reference 1 with "July 1975" and change "46" to "44"

2.) In Reference 2

a.) delete "Allis-Chalmer's Power Systems, Inc.,"

b.) replace "rpm" with "r/min",

c.) replace "46" with "44", and

d.) replace "[VL 04-0051540}" with "October 1975"

3.) In the Reference section of 10.2, replace References 7, 8, and 9 with the word "Deleted"

4.) In the Reference section of 10.2, replace the revision number of Reference 10 from "4" to "5", add "LP" after "Company," change the date from "December 15, 2003" to "January 18, 2007," and delete the VL number.

In the Reference section of 10.2, replace Reference 11 with the words "TP-03143, "Missile Analysis Methodology for GE Nuclear Steam Turbine Rotors by the SWPC," July 31, 2003."

LDCR-SA-2006-25, EVAL-2004-002055-01 (JCH):

Revise the second sentence in paragraph 10.2.2.6 to rep[lace "60 psig" with "psig"

Revise the next to last paragraph in 10.2.2.6.2.1 by replacing "seal oil storage tank" with "Hydrogen Degassing Storage Tank".

LDCR-SA-2006-25, EVAL-2004-002055-01 (JCH) (continued):

Revise the last sentence in the paragraph to clarify that the loop seal is permanently filled.

Revise the last paragraph to state "The seal oil drained from the seal oil tank passes into the Hydrogen Degassing Storage Tank to which the vapor exhauster is also connected. The exhauster creates a slight vacuum in the Hydrogen Degassing StorageTank so that oil saturated with hydrogen is degassed."

In Table 10.4-10, sheet 2 of 3, replace the current a. and b. item entries under "Generator Seal Oil Unit" with the following:

"Air Side Coolers", "2", "1", "113", and "0.65 x 10E6"

Revise the "Maximum Guaranteed Rating" and "Valves Wide Open" values for Hydrogen Pressure from 60 psig to 75 psig.

Revise the "service pressure" values for Hydrogen from "4.2 bar" aand "60 psig" to "5.1 bar" and "75 psig".

LDCR-SA-2008-8, EVAL-2007-002816-02 (RAS):

Revises position titles and organization descriptions to reflect approved titles maintained in EIS and the current CPNPP organizational structure.

LDCR-SA-2008-9, EVAL-2007-002223-01 (RJK):

CP License Amendment 128 changed the limit for identified primary to secondary leakage in TS LCO 3.4.13 to 150 GPD for any SG in each unit. This change removed an existing unit difference and brought the TS LCO into agreement with the standard technical specifications as modified by TSTF-449 R4. The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

FSAR Amendment 102, Supplement a

LDCR-SA-2008-14, EVAL-2008-002865-01 (TJEW):

Change title from "Director, Oversight & Nuclear Overview" to "Director, Oversight & Regulatory Affairs"

LDCR-SA-2008-15, EVAL-2008-002399-01 (JDS):

ANSI N512-1974 has been replaced with ASTM D 3912 Standard Test Method for Chemical Resistance of Coatings Used in Light-water Nuclear Power Plants which is endorsed by Reg guide 1.54 rev 1. The decontamination test per ANSI 512 was not carried into the replacement standard listed by the ASTM D33 committee that writes these procedures. FSAR Section 6.1B.2 was previously revised in Amendment 102 to remove ANSI N512 and update to current industry standards.

LDCR-SA-2007-15, EVAL-2005-004951-02 (JCH):

Section 10.4.2.2.1 System Operation : page 10.4-8

Deleted reference to priming tank.

Section 10.4.2.5 Instrumentation Requirements: page 10.4-9

Deleted reference to priming tank. Added description of the pressure switch that on low vacuum priming header pressure provides input to a main condenser priming trouble alarm. The main condenser vacuum priming trouble alarm is also activated if the main condenser outlet waterbox vacuum breaker air operated valve is not fully closed.

Justification: The priming tank was removed and a loop seal installed to prevent carry over of the water to the priming pumps due to repetitive reliability problems with the vacuum priming tank level control components.

LDCR-SA-2006-36, EVAL-2005-003364-19 (JDS):

Section 6.2.2 is updated for changes to the sump design and licensing basis to reflect the results of the mechanistic analysis requested in Generic Letter 2004-02. Section 6.2.2 References are administratively updated.

Section 6.2.2 is updated for changes to the sump design and licensing basis to reflect the results of the mechanistic analysis requested in Generic Letter 2004-02. Section 6.2.2.3.3 and 6.2.2.3.4 are revised to reflect the GSI-191 analysis and testing documented in SMF-2001-002201, SMF-2004-003972, and SMF-2005-003364 (see individual references and calculations above) as required by Generic Letter 2004-02. The change to the licensing basis is conservative in that analyses and testing have been performed to the current state of the art.

FSAR Amendment 102, Supplement a

LDCR-SA-2006-36, EVAL-2005-003364-19 (JDS) (continued):

Section 6.2.2 is updated for changes to the sump design and licensing basis to reflect the results of the mechanistic analysis requested in Generic Letter 2004-02. Table 6.2.2-4 is administratively updated per 10CFR50.71(e) to update and clarify the material description for the sump strainers per SFS-CP-GA-01.

Section 6.3.2.2.10 is updated for changes to the sump design and licensing basis to reflect the results of the mechanistic analysis requested in Generic Letter 2004-02. Section 6.3.2.2.10 is revised to reflect the NPSH analysis for RHR in ME(B)-325 R3.

Table 6.1B-1 is administratively updated per 10CFR50.71(e) to update the quantities of organic materials (cables and oil) from ACTN-MAN-2001-002201-72 and EE-CA-0000-5329.

LDCR-SA-2008-17, EVAL-2008-003064-02 (RAS):

Last sentence in section 6.4.5 incorrectly refers to the "Control Room Envelope Habitability Program" as the "Control Room Integrity Program". This change revises the sentence to use the correct program title.

LDCR-SA-2008-16, EVAL-2008-003163-02 (JCH):

Page 10.2-17 of Section 10.2.3.1, Materials Selection:

Changed compressive residual stress level from " shall not exceed 11,500 psi" to "shall not be lower than 14,504 psi (100 Mpa) and higher than 36,260 psi (250 Mpa) in the middle of the hub area; shall not be lower than 29,007 psi (200 Mpa) and higher than 58, 015 psi (Mpa) in the area of the rim of the hub; shall not be lower than 14, 504 psi (100 Mpa) and higher than 36,260 psi (250 Mpa) in the rim area.

Deleted " For up to 120 percent of rated speed the maximum disk stress at the shrink fit is less than half the burst strength of the material" and inserted " The maximum disk stress at the shrink fit is required to remain less than 50-60% of the yield strength of the material at nominal conditions."

Justification: New advanced design low pressure turbines were installed in Units 1 and 2. Advances in manufacturing techniques and material property selection have resulted in revision to the residual stess criteria. The new advanced design low pressure turbines incorporate disc which have these new advanced manufacturing techniques and new material compositions.

FSAR Amendment 102, Supplement a

LDCR-SA-2008-1, EVAL-2007-002946-02 (JDS):

FSAR changed to state, "Due to the sensitive nature of the seals, seal injection flow should be maintained when the Reactor Coolant System is not completely depressurized or when the Reactor Coolant System inventory is moving through the seal package." During outages seal injection flow has been secured and the RCPs are placed on their backseat which fulfills the intent of this section by isolating the RCP seals from the RCS. With RCPs not on a backseat during refueling, the FSAR was clarified to explain under what circumstances seal injection flow needs to be maintained to assure the RCP seal cleanliness.

FSAR Amendment 102, Supplement b

LDCR-SA-2008-2, EVAL-2006-003828-04 (GLM):

Due to degradation, non-safety vent chiller X-08 (CPX-CHCICE-08) is being replaced. The replacement is a York YK model higher pressure centrifigul liquid type ventilation chiller using R134a refrigerant. Since this chiller is a direct open versus hermetically sealed compressor a change to Section 9.4E.2 of the FSAR is necessary to reflect the modification changes in FDA-2006-3828-06.

Per FSAR Appendix 9.4E.1, the design function of the plant ventilation chilled water system is to remove heat rejected by plant equipment and to maintain ambient temperature below design limits within the areas it serves. The system is shown on Figure 9.4-11. The plant ventilation chilled water system is not required to mitigate the consequences of an accident or to maintain the plant in a safe shutdown condition. Except for the containment penetrations and containment isolation valves, the plant ventilation chilled water system is non-nuclear safety and non-seismic. By replacing chiller X-08 with a similar 660-ton chiller, the degraded system cooling capacity and efficiency is regained. The new chiller essentially provides the 660 ton desired cooling. This activity, therefore, improves the reliability of the ventilation chilled water system and has no adverse impact on the design function of the system.

LDCR-SA-2009-13, EVAL-2008-000762-02 (GLM):

In the 4th sentence from the bottom, change "Revision 1" to "Revision 0" for NUREG-700 (1981).

This is an editorial change. NUREG-0700, Revision 0 was issued in September 1981 and NUREG-0700, Revision 1 was issued in June 1996.

LDCR-SA-2009-8, EVAL-2009-000832-01 (JCH):

Section 10.2.2.10, Turbine Trips

Page 10.2-15: Delete the following: "The primary water pump shaft vibration trip shown in figure 10.2-1 is allowed to be defeated by the use of a time delay."

Justification: These trips are no longer automatic trips. They are manual. The automatic trips were removed with the installation of the new Turbine Protection System via FDA-2004-000773-01 (Unit 1) and FDA-2004-000773-02 (Unit 2).

Figure 10.2-1, "Turbine Trip Logic Diagram", needs to be revised to remove the logic for the following trips:

-Turb Gen Bearing Vibration (ETP)

-P.W. Pump Shaft Vibration

LDCR-SA-2009-8, EVAL-2009-000832-01 (JCH) (continued):

Justification: These trips are no longer automatic trips. They are manual. The automatic trips were removed with the installation of the new Turbine Protection System via FDA-2004-000773-01 (Unit 1) and FDA-2004-000773-02 (Unit 2).

LDCR-SA-2008-22, EVAL-2008-002039-02 (SCD):

Removes references to TLD readers from section 12.5.2.1. Changes Thermoluminescent dosimetry (TLD) to Optically Stimulated Luminescence (OSL) Badges, replaces TLD with OSL and replaces reference to onsite TLD reading with "The OSL Badges are processed and a definitive dose evaluation is provided by the approved service organization as requested." in section 12.5.2.5. Section 12.5.3.5 is changed to replace Thermoluminescent dosimetry (TLD) to Optically Stimulated Luminescence (OSL) Badges, replaces TLD with OSL and remove reference to onsite dosimetry reading. Requirement for offsite dosimetry service organization to maintain accreditation through the National Voluntary Laboratory Accreditation Program (NVLAP) is incoporated. Table 12.5-1 is changed to delete TLD Readers and Beta, Gamma, and Neutron Used for personnel dosimetry program.

LDCR-SA-2009-2, EVAL-2008-002473-02 (TJD):

Description of Change: Delete "the safety evaluations for" from FSAR section 17.2.1.1.3.1(2), which currently reads: "Review of the safety evaluations for: (1) procedures, (2) change to procedures,equipment, systems or facilities, and (3) tests or experiments where nuclear safety is adversely affected;"

Technical Justification: This change was originally approved as a part of LDCR SA-2000-23, but the current FSAR (Amendment 102) does not reflect the change. From a review of the current and previous FSAR versions, it appears that this portion of LDCR SA-2000-23 was inadvertently not incorporated into the FSAR. (Ref. correspondence, CPSES-200101291 and CPSES-200100873, for information on LDCR SA-2000-23.)

LDCR-SA-2009-6, EVAL-2004-001688-02 (JDS):

Document the use of the RHR pumps to fill the SI accumulators. The previously accepted method to fill and add water to the accumulators is by using the SI pump through the SI test header. When the RCS is less than 350 degrees F, TS 3.4.12 requires that the SI pumps be not "capable of injecting into the RCS." To accomplish accumulator fill with the SI pumps and meet the not "capable of injecting into the RCS" requires 40 fuses to be pulled to remove power to the valves on the SI test header. This method adds unnecessary challenges to equipment reliability (fuse clips) and increases the likelihood of human errors. Pulling 40 fuses is also manpower intensive, particularly during outage periods. Therefore, a method to fill and add water to the accumulator using RHR was implemented.

FSAR Amendment 102, Supplement b

LDCR-SA-2008-10, EVAL-2007-002677-04 (JCH):

Change Section 5.3.1.5 by adding "and 5.3.4-C" to first paragraph on page 5.3-5.

Justification: Changes reflect the use of a hydraulic nut closure system. The HydraNut is a hydraulically actuated, self-tensioning nut. The result of the installation will be a set of HydraNuts that tension the reactor vessel studs in one operation using hydraulic pressure, as opposed to several passes using stud tensioners currently required for the conventional nuts. As a result, the HydraNut will significantly decrease the time required to tension the reactor vessel studs.

Changed section 5.3.1.7 first paragraph to add the following statement: "Conventional nuts and washers were originally supplied with the reactor vessel. Hydraulic nuts were procured to be used with the reactor vessel studs as appropriate."

Added reference to Table 5.3-4C at the end of paragraph 2.

Justification: Changes reflect the use of a hydraulic nut closure system. The HydraNut is a hydraulically actuated, self-tensioning nut. The result of the installation will be a set of HydraNuts that tension the reactor vessel studs in one operation using hydraulic pressure, as opposed to several passes using stud tensioners currently required for the conventional nuts. As a result, the HydraNut will significantly decrease the time required to tension the reactor vessel studs.

Added new Table 5.3-4C, Unit 1 and 2 Reactor Vessel Closure Bolting Material Properties - Hydranuts & Washers.

Justification: Changes reflect the use of a hydraulic nut closure system. The HydraNut is a hydraulically actuated, self-tensioning nut. The result of the installation will be a set of HydraNuts that tension the reactor vessel studs in one operation using hydraulic pressure, as opposed to several passes using stud tensioners currently required for the conventional nuts. As a result, the HydraNut will significantly decrease the time required to tension the reactor vessel studs.

LDCR-SA-2009-11, EVAL-2009-001859-02 (JDS):

Delete inserts and adaptors from under Closure studs, nuts, washers due to implementation of reactor vessel hydraulic nuts (HydraNuts).

Delete as modified by Code Case 1605 from under SA-540 Class-3 B24. Code case 1605 is not applicable to the Comanche Peak Unit 1 and 2 reactor vessel studs, nuts and washers. The chemistry requirements and increased material size limit for SA-540 Grade B24 bolting material referenced in Code Case 1605 does not apply to the Comanche Peak reactor vessel studs, nuts and washers. The Code Case is also not applicable to the new HydraNut reactor vessel modification.

LDCR-SA-2009-11, EVAL-2009-001859-02 (JDS) (continued):

Delete Code Case 1605. Code case,1605 is not applicable to the Comanche Peak Unit 1 and 2 reactor vessel studs, nuts and washers. The chemistry requirements and increased material size limit for SA-540 Grade B24 bolting material referenced in Code Case 1605 does not apply to the Comanche Peak reactor vessel studs, nuts and washers. The Code Case is also not applicable to the new HydraNut reactor vessel modification.

LDCR-SA-2008-23, EVAL-2008-000624-02 (TJD):

The following sentence is being deleted from FSAR, section 17.2.17, at the top of page 17.2-26: "The Nuclear Overview Department reviews and approves the administrative control procedures and instructions and the retention periods assigned for quality assurance records." This change will make section 17.2.17 consistent with FSAR Section 17.2. Also, FSAR section 17.2 pages 17-ii and 17.2-29 are being revised to reflect current organizational titles.

The review of procedures for quality assurance requirements is now accomplished via the SORC process, as identified in FSAR Section 17.2.2. The change to the SORC process was a reduction in commitment which was reviewed and approved by the NRC via LDCR SA-89-243, therefore; LDCR SA-2008-023 does not constitute a reduction in commitment. Commitment 22910 addresses this review process. Additionally, ACTN-MAN-2008-000624-02 has been created to delete commitment 22906, which discusses the pre-SORC review process and should have been deleted when the current review process was approved by the NRC.

FSAR Amendment 103

LDCR-SA-2004-26, EVAL-2003-000239-03 (SCD):

EVAL-2003-000239-03 (LDCR-SA-2004-026): Corrects note reference for the Heater Drain Tank Equalization Header from "**" to "*". This is an editorial change.

EVAL-2003-000239-03 (LDCR-SA-2004-026): Adds Note 1 to Table 10.4-20 to describe the use of the 2 pH sample points.

EVAL-2003-000239-03 (LDCR-SA-2004-026): Delete dissolved oxygen and pH monitoring from various sample points as identified in FSAR Table 10.4-20. This activity provides new and enhanced models of the existing oxygen and pH analyzers. The Secondary Chemistry Control Program monitoring capabilities will continue to meet the EPRI "PWR Secondary Water Chemistry Guidelines."

LDCR-SA-2009-5, EVAL-2009-000834-02 (CBC):

TMI Section I.A.1.3, "Shift Manning," is updated to reflect the work hour requirements prescribed by 10 CFR Part 26, Subpart I, effective 10/01/2009. Work hour restrictions are being deleted from Technical Specification 5.2.2.d (License Amendment Request 09-006) to support compliance with 10 CFR Part 26. This change will make the FSAR consistent with the Technical Specifications and the new rule. (EVAL-2009-000834-01).

LDCR-SA-2009-7, EVAL-2008-002474-05 (TJD):

Revise the title of FSAR, section 17.2.7.1, from "Receipt Inspection" to "Receipt Inspection and Material Testing."

This change will clearly define the role of the Procurement Quality Assurance Receipt Inspection as "Receipt Inspection and Material Testing."

Revise the first sentence in FSAR, section 17.2.7.1, from "Receipt inspections at CPSES are . . ." to "Receipt inspection and material testing at CPNPP is . . ."

This change will clearly define the role of the Procurement Quality Assurance Receipt Inspection as "Receipt Inspection and Material Testing."

Change FSAR 17.2.7.1, item 2, from CPSES to CPNPP.

Update plant name.

Change FSAR, section 17.2.10, second sentence, from ". . . and 1.116 as . . ." to ". . . 1.116, and 1.123 as . . ."

This change updates the FSAR to include a Regulatory Guide (1.123) that we are currently commited to follow.

Update two places on FSAR page 17.2-20 from CPSES to CPNPP.

LDCR-SA-2009-7, EVAL-2008-002474-05 (TJD) (continued):

Update plant name.

Page 17.2-20, Change the wording in the next to last paragraph from ". . . level III inspector . . ." to ". . . Level II or III inspector . . ."

Revise the FSAR to allow a Level II inspector to assign hold points and ensure that inspection methods are adequate.

LDCR-SA-2009-17, EVAL-2007-001302-01 (TJD):

Revise the Discussion section for Regulatory Guide 1.33 to include an additional exception for commercial grade calibrations.

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

Revise the Discussion section for Regulatory Guide 1.123 to include an additional exception for commercial grade calibrations.

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

Revise the Discussion section for Regulatory Guide 1.144 to include an additional exception for NVLAP or A2LA accredited commercial grade calibrations.

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

Revise FSAR section 17.2.4 to describe the procurement document control requirements needed to support the use of NVLAP or A2LA accredited commercial grade calibration services.

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

Revise FSAR section 17.2.7 to describe changes to the control of purchased safety-related material, equipment and services needed to support the use of NVLAP or A2LA accredited commercial grade calibration services.

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

Revise FSAR section 17.2.18.1, "Audit Organizations," to explain the audit implications when using NVLAP or A2LA accredited commercial grade calibration services.

FSAR Amendment 103

LDCR-SA-2009-10, EVAL-2005-000768-02 (TJD):

This change supports a new option for NVLAP or A2LA accredidation of commercial grade calibration services in lieu of supplier audit by licensee.

FSAR Table 2.5.6-7, Station 46+17, the Type should be changed from "Pneumatic" to "Pneumatic (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 55+95, the Type should be changed from "Pneumatic" to "Pneumatic (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 56+05, the Type should be changed from "Pneumatic" to "Wellpoint (Replaced)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 56+05, the Piezometer No. should be changed from "P-II-1a" to "P-II-1" The "a" needs to be removed from the piezometer number.

This change corrects the SCR dam piezometer number.

FSAR Table 2.5.6-7, Station 56+02, the Type should be changed from "Wellpoint" to "Wellpoint (Replaced)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 56+22, the Type should be changed from "Wellpoint" to "Wellpoint (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 56+92, the Type should be changed from "Wellpoint" to "Wellpoint (Replaced)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 56+06, the Type should be changed from "Pneumatic" to "Pneumatic (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 69+00, the Type should be changed from "Wellpoint" to "Wellpoint (Replaced)"

LDCR-SA-2009-10, EVAL-2005-000768-02 (TJD) (continued):

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 69+05, the Type should be changed from "Wellpoint" to "Wellpoint (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 69+15, the Type should be changed from "Wellpoint" to "Wellpoint (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 69+09, the Type should be changed from "Pneumatic" to "Pneumatic (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Table 2.5.6-7, Station 69+20, the Type should be changed from "Pneumatic" to "Pneumatic (Abandoned)"

This change updates the table to reflect changes to the SCR dam piezometers.

FSAR Figure 2.5.6-56 is updated to reflect the same piezometer changes that were made to FSAR Table 2.5.6-7, as detailed in the previous 13 records of this database.

This change updates Figure 2.5.6-56 to reflect changes to the SCR dam piezometers.

LDCR-SA-2009-16, EVAL-2009-003424-01 (TJEW):

On Table 8.3-3, sheet 1, for items:

3A, revise the words, "loss of power to buses 1EA1 and 1EA2 from XST1"

to read, "For open circuit: Loss of alternate power to buses 1EA1 and 1EA2 from XST1.

For short circuit: Loss of power to buses 2EA1, 2EA2, and loss of alternate power source for buses 1EA1 and 1EA2."

3B, revise the words, "loss of power to buses 2EA1 and 2EA2 from XST1"

to read, "For open circuit: Loss of power to buses 2EA1 and 2EA2 from XST1.

LDCR-SA-2009-16, EVAL-2009-003424-01 (TJEW) (continued):

For short circuit: Loss of power to buses 2EA1, 2EA2, and loss of alternate power source for buses 1EA1 and 1EA2."

On Table 8.3-3, sheet 2, for items:

4A, revise the words, "loss of power to buses 1EA1 and 1EA2 from XST2"

to read, "For open circuit: Loss of power to buses 1EA1 and 1EA2 from XST2.

For short circuit: Loss of power to buses 1EA1, 1EA2, and loss of alternate power source for buses 2EA1 and 2EA2.”

4B, revise the words, "loss of power to buses 2EA1 and 2EA2 from XST2"

to read, "For open circuit: Loss of alternate power to buses 2EA1 and 2EA2 from XST2.

For short circuit: Loss of power to buses 1EA1, 1EA2, and loss of alternate power source for buses 2EA1 and 2EA2."

LDCR-SA-2009-3, CR-EVAL-2009-000413-01 (SCD):

Changes to CPNPP Executive Management assignments and Maintenance Department reorganization.

LDCR-SA-2009-23, EVAL-2006-003201-02 (TJEW):

The revised duty cycle load of the batteries envelops the calculated duty cycle load of the batteries and will maintain the end of the duty cycle battery voltage of 1.75 VPC = battery voltage of 105V, as required by Section 4.1.2.1.B of DBD-EE-044 Rev 24 and FSAR section 8.3.2.1 item "2" Capacity sub item "a" Batteries. Ref calculation EE-CA-0000-5454 Rev 0. The listing of enveloping duty cycle load capability of battery in Table 8.3-4 will minimize unnecessary FSAR changes when a battery load changes but remain bounded by the enveloping load. 59SC-2006-003201-01-00 provide the basis that a 50.59 evaluation is not required for this change.

FSAR Table 8.3-4 is revised to list the duty cycle load of batteries which envelop the batteries calculated load and maintains end of the duty cycle voltage of 105V.

FSAR Amendment 103

LDCR-SA-2008-11, CR-EV-2006-003080-00-150 (JCH):

Table 10.4-10: Maximum System Heat Sink Requirements

1.Exciter air coolers: The turbine generator exciter air coolers currently have a heat load of 2.143x10^6 Btu/hr, this will be increased to 2.184x10^6 Btu/hr for SPU. Additionally, the exciter air coolers currently have a cooling water flow rate of 790 gpm, this will be increased to 792.5 gpm for SPU.

LDCR-SA-2008-11, CR-EV-2006-003080-00-150 (JCH) (continued):

2.Hydrogen coolers: The turbine generator Hydrogen coolers currently have a heat load of 17.02x10^6 Btu/hr, this will be decreased to 16.79x10^6 Btu/hr for SPU. Additionally, the Hydrogen coolers have a cooling water flow rate of 4480 gpm, this will be decreased to 4403 gpm for SPU.

3. Isophase forced cooling unit: The isophase bus forced cooling units currently have a heat load of 1.03x10^6 Btu/hr, this will be increased to 1.808x10^6 Btu/hr for SPU. Additionally, the isophase forced cooling units have a cooling water flow rate of 137 gpm, this will be increased to 220 gpm for SPU.

4. Primary water coolers: The turbine generator primary water coolers currently have a heat load of 31.17x10^6 Btu/hr, this will be increased to 32.70x10^6 Btu/hr for SPU. Additionally, the primary water coolers have a cooling water flow rate of 5720 gpm, this will be increased to 7115 gpm for SPU. To accomplish this five of six cooler will be placed in operation for SPU, currently four of six coolers are in operation.

Justification:

Heat load, flow, and pressure requirements for the exciter air coolers, Hydrogen coolers, and primary water coolers can be found in Vendor Document EC-06153 "Thermal Uprate Study 4.5% for TXU Comanche Peak Units 1 and 2" (Ref ENR-2008-003080-12). Heat load for these coolers is calculated using conservative assumptions, each cooler has its own set of conservative conditions.

Heat load and flow requirements for the isophase bus forced cooling units can be found in Section 3.13.1 of ER-SPU-4.2.6 "Turbine Plant Cooling Water System" (Ref ENR-2008-003080-39).

These changes are required to maintain configuration control.

Section 10.2.2.6.1: Hydrogen Storage and Supply

Hydrogen pressure will be increased to 65 psig for SPU, currently pressure is 60 psig.

LDCR-SA-2008-11, CR-EV-2006-003080-00-150 (JCH) (continued):

Justification:

Heat load, flow, and pressure requirements for the exciter air coolers, Hydrogen coolers, and primary water coolers can be found in Vendor Document EC-06153 "Thermal Uprate Study 4.5% for TXU Comanche Peak Units 1 and 2" (Ref ENR-2008-003080-12). Heat load for these coolers is calculated using conservative assumptions, each cooler has its own set of conservative conditions.

Heat load and flow requirements for the isophase bus forced cooling units can be found in Section 3.13.1 of ER-SPU-4.2.6 "Turbine Plant Cooling Water System" (Ref ENR-2008-003080-39).

These changes are required to maintain configuration control.

Section 10.4.12.1: Design Basis

The TPCW system currently has a maximum operating heat exchanger outlet temperature of 105F, this will be increased to 107F for SPU. Currently the circulating water temperature that flows through the tube side of the TPCW heat exchanger is 95F, this will be increased to 102F post-SPU.

Justification:

Heat load, flow, and pressure requirements for the exciter air coolers, Hydrogen coolers, and primary water coolers can be found in Vendor Document EC-06153 "Thermal Uprate Study 4.5% for TXU Comanche Peak Units 1 and 2" (Ref ENR-2008-003080-12). Heat load for these coolers is calculated using conservative assumptions, each cooler has its own set of conservative conditions.

Heat load and flow requirements for the isophase bus forced cooling units can be found in Section 3.13.1 of ER-SPU-4.2.6 "Turbine Plant Cooling Water System" (Ref ENR-2008-003080-39).

These changes are required to maintain configuration control.

LDCR-SA-2008-12, CR-EV-2006-003080-00-103 (JCH):

Section 10.4.10.5 : remove the following "Heater drain tank control valves to heaters 3A and 3B are interlocked to close when the corresponding extraction steam motorized stop valve is colsed. Note however that Operations may set vent valves in the OPEN position by pulling their fuses. The Heater Drain System Operating Procedures will control the valves configuration."

LDCR-SA-2008-12, CR-EV-2006-003080-00-103 (JCH) (continued):

Justification: The 3A and 3B heater drain tank control vent valves are being removed and replaced with straight pipe to allow free flow of steam from the heater drain tanks to the 3A and 3B heaters. This will allow the heater drain system to vent with less restriction and better respond to plant trainsients after the power uprate. The valves are currently locked open and act as an oriface or restriction in the line impeeding proper steam vent flow.

Section 10.4.11.5: remove the following "Heater drain tank control vent valves to heater 3A and 3B are interlocked to close when the corresponding extraction steam motorized stop valve is closed. Note the vent valves may be disabled in the open position by removing the control fuses and/or isolating the positioner instrument air supply and restraining the valve handwheels. The vent globe valves are air operated and fail open"

Justification: The 3A and 3B heater drain tank control vent valves are being removed and replaced with straight pipe to allow free flow of steam from the heater drain tanks to the 3A and 3B heaters. This will allow the heater drain system to vent with less restriction and better respond to plant trainsients after the power uprate. The valves are currently locked open and act as an oriface or restriction in the line impeeding proper steam vent flow.

LDCR-SA-2008-20, CR-EV-2007-002129-00-4 (JCH):

Section 13.3B.1.1 Plant Manager:

Revised Plant Manager responsibilities to be maintenance and implementation of the Fire Protection Progrram, not development.

Justification: This change is required due to various personnel and organizational changes as CPNPP. This change is administrarive in nature and does not change the Fire Protection Program requirements at CPNPP.

13.3B.1.2 Director, Maintenance:

Director, Maintenance responsible for maintenance and implementation of the Fire Protection Program, not implementation.

The Manager, Technical Support does not report to him so the reference to him was deleted.

13.B.1.3 Director, Site Engineering:

Added new section for Director Site, Engineering who is responsible for the development of the Fire Protection Program. This responsibility is assigned to the Manager, Technical Suport.

LDCR-SA-2008-20, CR-EV-2007-002129-00-4 (JCH) (continued):

13.B.1.4 Manager, Technical Support:

Manager, Technical Support is responsible to the Director, Site Engineering instead of the Director, Maintenance.

Added the responsibility for "review and apporval of Fire Protection Design documents, licensing documents, and Station Procedures including revisions, to verity technical accuracy and regulatory compliance.

Added the following paragraph- "The Manager, System Engineering is responsible to the Director, Site Engineering for technical support of the Fire Protection Program in relation to the review and implementation of design modifications and the resolution of system issues.

13.3B.1.5 Maintenance Team Manager:

Deleted "Smart" from Maintenance Team Manager title.

Added responsibility "to assist the Nuclear Training Manager in the development and implementation of fire protection training programs for personnel and the fire brigade at CPNPP.

Justification: This change is required due to various personnel and organizational changes as CPNPP. This change is administrarive in nature and does not change the Fire Protection Program requirements at CPNPP.

LDCR-SA-2008-13, CR-EV-2006-003080-94 (JDS):

Incorporate Westinghouse Topical Reports associated with conversion to the Westinghouse transient and accident analysis methodologies.

Include relevant information resulting from the removal of Appendix 4.A and 4.B into this section for consistency with other Westinghouse plants using the Westinghouse transient and accident analysis methodologies.

Remove Appendix 4.A and incorporate the relevant information into chapters 4.1, 4.2, 4.3, and 4.4 for consistency with other Westinghouse plants using the Westinghouse transient and accident analysis methodologies.

Update Chapter 6.2 to reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Insert Tables to reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

LDCR-SA-2008-13, CR-EV-2006-003080-94 (JDS) (continued):

Replace Tables to reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Remove Table as table does not reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Modify Table to reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Delete Tables which does not reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Insert Figures to reflect GOTHIC code analysis provided to the NRC for containment accident conditions resulting from 4.5% power uprate of Units 1 and 2.

Describe Power Distribution Monitoring System Using BEACON.

Update time required for boric acid solution to counteract xenon decay when charging to the RCS is via reactor coolant

pump seal injection at the rate of approximately 5 gpm per pump from approximately 3.5 hours to 6 hours. This is consistent with the Westinghouse Methodology.

Revise this section for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Revise these Tables for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Replace this figure for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Insert these Figures for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Revise these sections for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Insert these Tables for consistency with Westinghouse transient and accident analysis methodologies now used on Units 1 and 2.

Update Table on Reactor Design Comparison for consistency with uprated conditions of Units 1 and 2.

LDCR-SA-2008-13, CR-EV-2006-003080-94 (JDS) (continued):

Update Table on Analytical Techniques in reactor design for consistency with Westinghouse Methodologies used.

Insert Table on Thermal and Hydraulic Comparison for pre- and post- uprate conditions of Units 1 and 2.

LDCR-SA-2009-19, CR-EV-2009-000214-11 (JDS):

Update Chapter 9 to modify the time required to add enough boric acid solution to bring the plant to hot shutdown and counteract xenon decay if normal charging is unavailable.

Update Chapter 4 to include a description of the Westinghouse Integrated Nozzle (WIN) top nozzle design. The mechanical design evaluations idocument that the new fuel assemblies are compatible with existing fuel, reactor vessel internals, spent fuel racks, fuel handling equipment and RCCAs and that all structural integrity and seismic requirements are satisfied.

LDCR-SA-2007-10, CR-EV-2006-003080-202 (JDS):

Revise the Main Generator rating from 1350 MV to 1410 MVA for each Unit in the first sentence of the second paragraph insection 1.2.2.5.

Revised the titles for Figures 10.1-1 and 10.1-2 as follows:

10.1-1 Unit 1, 100% Power Heat Balance (See VL-07-001239)

10.1-2 Unit 2, 100% Power Heat Balance (See VL-07-001240)

The existing figures are replaced with new heat balance drawings supplied by Siemens. These figures contain information proprietary to Siemens. Consequently, no figures will be included but the List of Figures is revised to refer to the vendor control number for the drawings which can be accessed via the plant record system.

Revised the sixth paragraph in section 10.1 as follows:

"Heat balances reflecting the design at an NSSS rated thermal power of 3628 MWt (maximum guaranteed rating) are shown on Figures 10.1-1 and 10.1-2, respectively. The rating of the electric generator is 1,410,000 kVA, 60 Hz, 0.90 power factor."

Replace the 2nd and 3rd paragraphs with the following:

"The Unit 1 turbine generator is designed to produce 1263.4 MWe when operating at 14,774,975 lb/hr saturated steam at 980 psia and 1192.3 Btu/lb with 0.21% moisture at the throttle and 1.414 in Hg absolute at the turbine exhaust. The Unit 2 turbine generator is idesigned to produce 1253.4 MWe when operating at 14,715,715 lb/hr saturated steam

LDCR-SA-2007-10, CR-EV-2006-003080-202 (JDS) (continued):

at 948.9 psia and 1192.1 Btu/lb with 0.40% moisture at the throttle and 1.414 in Hg absolute at the turbine exhaust."

Revise the main generator kVA rating in the first sentence in section 10.2.2.6 by replacing "1,350,000" with "1,410,000" and deleting "and a short circuit ratio of 0.58."

In section 10.2.2.6.1, reevise the maximum pressure in the second sentence from "60" psig to "75" psig.

In section 10.4.1.1.1, revise the value for:

Exhaust steam from "8 x10E6" to "8.44x10E5"

Condenser duty from "7.7x10E9" to "8.06x10E9"

Condenser pressure at turbine exhaust from "3.5" to "3.55"

Outlet temperature form "110" to "110.8".

In section 10.4.5.1, in the third sentence, revise the value for total heat removed from "8.4X10E9" to "8.8x10E9" and amount removed from the main cindenser from "8.0x10E9" to "8.4x10E9".

Delete Figures 10.1-1 and 10.1-2 for Unit 1 100% Power Heat Balance and Unit 2 100% Power Heat Balance

Replace existing Table 10.1-1 with new Table reflecting new values for Maximum Guaranteed Rating for Unit1 and for Unit 2. Delete the column for "Valves Wide Open" values.

LDCR-SA-2004-18, EVAL-2003-000167-02 (JDS):

This Activity installs neutron dosimetry sensors used to monitor neutron exposure for continued surveillance of reactor vessel integrity. The equipment is installed in the reactor cavity annulus in all four quadrants utilizing the neutron detector well ventilation ducts for support.

By installing the Ex-Vessel Dosimetry, an alternative dosimetry program is instituted to monitor reactor vessel neutron exposure through the license renewal period. Therefore, CPSES is following the guidance for reactor vessel surveillance as stated in the NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," April 2001.

FSAR Amendment 103a

LDCR-SA-2009-12, EV-CR-2008-003459-00-11 (SCD):

Amend UFSAR Section 6.2.4.1.3.1 to change the containment isolation valve typical arrangements applied to the high head safety injection lines and to describe that pressure indication is provided to monitor ECCS piping pressure.

Add Section 6.3.5.2.6 and REFERENCES to the Table Of Contents, add Section 6.3.5.2.6 to describe CPNPP response to GL 2008-01, describe the new ECCS piping pressure monitoring function and add Reference section with GL 2008-01 as a reference.

Amend UFSAR Figure 6.2.4-1 Valve Arrangement 9 to depict the locked closed manual containment isolation valve and downstream pressure transmitter.

Amend UFSAR Figure 6.2.4-1 to add new Valve Arrangement 46.

Amend UFSAR Section 5.2.5.2.3 to describe that pressure indication is provided to monitor ECCS piping pressure.

LDCR-SA-2008-19, EV-CR-2007-002125-00-2 (JCH):

13.3.B.5 QUALITY ASSURANCE PROGRAM

1. Design Control and Procurement Document Control

Changed first paragraph to read as follows:

"Measures will be established to assure that NRC applicable guidelines or NRC approved alternatives are included in design and procurement documents and that any deviations therefrom are controlled. There procedures include provisions, as necessary, to ensure that:"

4. Inspection

Changed to read as follows:

"A program for independent inspection of activities affecting fire protection will be established and executed by, or for, the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activity."

Justification:

The proposed changes revise the language in FSAR 13.3B.5 to reflect the guidance language in Appendix A to BTP APCSB 9.5-1. CPSES is committed to the meet the requirements of Appendix A to BTP APCSB 9.5-1 as stated in NRC SSERs 1, 12, and 21.

The changes will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

LDCR-SA-2008-19, EV-CR-2007-002125-00-2 (JCH) (continued):

13.3.B.5 QUALITY ASSURANCE PROGRAM

5. Test and Test Control

Added the following sentence:

"Test results will be properly evaluated and acted on.

7. Nonconforming Items.

Changed to read as follows:

"Measures are established to control items that do not conform to specified requirements to prevent inadvertent use of installation."

8. Corrective Action

Changed to read as follows:

"Measures will be established to ensure that conditions adverse to fire protection such as failures, malfunctions, deficiencies or deviations, defective components, uncontrolled combustible material and non-conformances are promptly identified, reported, and corrected."

10. Evaluations

Changed heading to "Audits".

Changed to read:

"Audits will be conducted and documented to verify compliance with the fire protection program, including design and procurement documents, instructions, procedures and drawings, and inspection and test activities. These audits will be performed by personnel not having direct responsibility for the Fire Protection Program."

Justification:

The proposed changes revise the language in FSAR 13.3B.5 to reflect the guidance language in Appendix A to BTP APCSB 9.5-1. CPSES is committed to the meet the requirements of Appendix A to BTP APCSB 9.5-1 as stated in NRC SSERs 1, 12, and 21.

The changes will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

FSAR Amendment 103a

LDCR-SA-2010-2, EV-CR-2005-003122-00-4 (JCH):

6. Inspection, Test and Operating Status.

Change to read "Measures will be established to provide for the identification of items that have satisfactorily passed required tests and inspections.

Justification:

FSAR Section 13.3B.5.6 should reflect what is stated in C.6 of Appendix A to BTP APCSB 9-5.1 and should not contain the language related to tagging of non-conformances. Non-conformances are discussed in FSAR 13.3B5.7. As presently worded this section can lead to unintended and overly restrictive requirements placed on the Fire Protection Program. This change will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

LDCR-SA-2009-24, EV-CR-2006-004121-00-1 (JDS):

Add a paragraph to document the use of the demineralized water storage tank and transfer pump to manually makeup to the containment spray pump suction piping located downstream of the containment sump isolation valves. The change allows for the fill of a section of containment spray pump suction piping with demineralized water, rather than borated water. This method is also used to fill the containment spray risers in containment.

LDCR-SA-2010-11, EV-CR-2010-001826-00-5(TJD):

Delete all the text in the Discussion section of Regulatory Guide 1.16, and replace it with the following: "This regulatory guide was withdrawn on August 11, 2009."

Technical Justification: Regulatory Guide 1.16 was withdrawn by the NRC on August 11, 2009 (Ref. Federal Register, 74 FR 40244, 8/11/2009).

Delete the following sentence from the CPSES Response portion of FSAR section II.K.3.3, "All challenges to SV or RV shall be documented in the monthly operating report." Then replace the deleted sentence with the following: "The reporting of SV & RV challenges was moved from the annual report to the monthly operating report (TS 6.9.1.5), then License Amendment 64 deleted this reporting requirement altogether."

Technical Justification: License Amendment 64 was implemented in July 1999 and deleted TS 6.9.1.5, including the requirement to report challenges to PORVs and safety valves, which was being reported in the annual operating report.

FSAR Amendment 103a

LDCR-SA-2009-25, EV-CR-2007-002527-00-2 (TJEW):

Changes to the AAP are controlled by the site facility program (reference STA-716) and are not required to be tracked as plant equipment. Therefore, the current AAP configuration will not be tracked on plant drawing, and drawing M1-0313 Sht A Sht - will be retired. This activity has no adverse impact on the design function of any SSC because the drawings that are being retired reflect a component was not implemented in the field and have inaccurate depictions of the current AAP status.

Delete the reference to M1-0310-A on Sheet 12 of Table 3.2-3

Changes to the AAP are controlled by the site facility program (reference STA-716) and are not required to be tracked as plant equipment. Therefore, the current AAP configuration will not be tracked on plant drawing, and drawing M1-0313 Sht A Sht - will be retired. This activity has no adverse impact on the design function of any SSC because the drawings that are being retired reflect a component was not implemented in the field and have inaccurate depictions of the current AAP status.

Delete the reference to M1-0310-A on Sheet 10 of Table 3.2-4

Changes to the AAP are controlled by the site facility program (reference STA-716) and are not required to be tracked as plant equipment. Therefore, the current AAP configuration will not be tracked on plant drawing, and drawing M1-0313 Sht A Sht - will be retired. This activity has no adverse impact on the design function of any SSC because the drawings that are being retired reflect a component was not implemented in the field and have inaccurate depictions of the current AAP status.

Delete the reference to M1-0310-A on page 9-xv

LDCR-SA-2009-21, EV-CR-2009-000214-00-8 (JDS):

The misloaded assembly analysis methodology described in the FSAR is being changed from the Luminant Power methods detailed in topical report RXE-91-002 to the original Westinghouse methodology previously reviewed and approved for Comanche Peak by the NRC (NUREG-0797, Section 15.3.3). This change is necessary to fully complete the transition to the Westinghouse accident analysis methodology. An improved description of the original Westinghouse methodology has been recently provided in WCAP-16676-NP which is also referenced as part of the adoption of the misloaded assembly analysis methodology. The improved description (WCAP-16676-NP) provides additional clarifying details regarding the original methodology but does not modify or alter the method relative to the original Westinghouse methodology approved by the NRC for Comanche Peak (NUREG-0797, Section 15.3.3). The specific application of the methodology is to analyze reactor operation with a fuel assembly inadvertently loaded into an improper location. Luminant Power is using the original analysis methodology approved for Comanche Peak in the specific application as approved by the NRC. The SER did not require any additional restrictions or limitations beyond those described in the methodology.

Update table of contents to reflect revised section.

FSAR Amendment 103a

LDCR-SA-2010-10, EV-CR-2009-000465-00-3 (JDS):

Recently, Westinghouse determined that existing Rod Withdrawal at Power (RWAP) RCS overpressure analyses do not cover the full range of reactor power operations for all Westinghouse-designed PWRs (W-PWRs).

Westinghouse performed generic RWAP analyses to demonstrate that the peak RCS pressure criterion was met for most W-PWRs equipped with a positive flux rate reactor trip (PFRT).

Additionally, an evaluation was performed against a generic analysis for Comanche Peak Units 1 and 2. The evaluation against the generic RWAP overpressure analyses concluded that credit for the PFRT was sufficient to demonstrate compliance with the RCS overpressure criterion. Westinghouse has performed analyses for various plants to conservatively calculate Reactor Coolant System (RCS) pressure following a Rod Withdrawal at Power (RWAP) event. Specifically, the Westinghouse analysis for Comanche Peak Units 1 and 2 calculated an acceptable result for RCS peak pressure.

Comanche Peak has reviewed the information and this FSAR update captures the appropriate information from the evaluation regarding the safety analysis by adding the Uncontrolled RCCA Bank Withdrawal to the Power Range high positive neutron flux rate trip discussion.

Comanche Peak has reviewed the information and this FSAR update captures the appropriate information from the evaluation regarding the safety analysis by inserting Power range high flux rate in the uncontrolled rod cluster control assembly bank withdrawal at power.

Recently, Westinghouse determined that existing Rod Withdrawal at Power (RWAP) RCS overpressure analyses do not cover the full range of reactor power operations for all Westinghouse-designed PWRs (W-PWRs).

Comanche Peak has reviewed the information and this FSAR update captures the appropriate information from the evaluation regarding the safety analysis by updationg the reactor protection system automatic features with a discussion of a trip when any two of four power range neutron flux high positive rate channels exceed a setpoint.

FSAR Amendment 103b

LDCR-SA-2010-6, EV-CR-2003-002117-00-2 (JCH):

Section 10.4, Table 10.4-13 (Auxiliary Steam System Requirements)

Description of Change:

Modify the entry stating that the Turbine Gland Steam System is only required by the Feedwater Pump Turbine during startup only to add "Power Operations: 230 (115 Each) 60-150 psig SAT".

Technical Justification:

The Auxiliary Steam System supplies 150 psig steam to the Feedwater Pump Turbine (FWPT) Gland Steam (GS) supply valves (u-PV-3918A). The GS supply valves reduce the 150 psig Auxiliary Steam (AS) supply to 2-4 psig. Each GS supply valve supplies both FWPTs for the associated unit.

This LDCR will change FSAR Table 10.4-13 to agree with the current FSAR description of the AS/GS system operation.

The FSAR Turbine GS System description states the following (10.4.3.2(1.b) :

“The pressure at the high-pressure end is insufficient to seal the low-pressure seals. Auxiliary steam from the Auxiliary Steam System is provided at the inner annular chambers through the steam seal supply valve.”

The FSAR Turbine GS System design bases states the following (10.4.3.1(2) :

“The seal steam supply header has a normal flow rate of 230 lb/hr at 2 to 4 psig. At startup, 1150 lb/hr of auxiliary steam from the Auxiliary Steam System is supplied at 2 to 4 psig.”

This indicates that the FWPT seals are not self-sealing, and the flow rate of the seal steam will vary with loading, from 1150 lb/hr at startup to 230 lb/hr during normal plant operations.

LDCR-SA-2009-9, EV-CR-2007-001888-00-6 (RAS):

Insert a parenthetical reference to Fig. 1.2-8 and 1.2-15 in the 1st sentence of the 3rd paragraph to clarify the physical location of the Jib Crane on the 905 level of the Containment.

In the 4th sentence of the 3rd paragraph, replace "Crane" with "telescopic jib crane" to clarify the particular crane of interest.

LDCR-SA-2009-9, EV-CR-2007-001888-00-6 (RAS) (continued):

In the last sentence of the 3rd paragraph, replace "jib" with "The telescopic jib" to clarify the particular crane of interest

Insert the following immediatley after the 3rd paragraph to describe the administrative controls applicable to Jib Crane operation:

"Both the Polar Crane and the Telescopic Jib Crane operating at the same time constitutes Multi-crane Operation which requires special administrative controls to reduce the likelihood of unplanned crane interactions. These administrative controls include a qualified multi-crane coordinator located in a position of good visibility to opbserve both cranes."

Revise the first four sentences of the last paragraph as follows:

"A Containment Anti-Collision Control System is also installed on the telescopic jib crane to reduce the likelihood of unplanned crane interactions with the polar crane bridge. The function of the Progammable Logic Controller (PLC) and associated components is to keep the jib crane boom below the Polar Crane bridge. In addition, an administratively controlled key, which is normally placed in the Polar Crane allows full operation from the polar crane radio controller. In this mode without the key, the PLC restricts the height of the Jib Crane boom."

Delete the following last sentence from the last paragraph:

"These administrative controls include a multi-crane coordinator located in a position of good visibility to observe both cranes."

Insert a reference to Figure 9.1-14 at the end of the first paragraph.

Insert the following sentence at the end of the second paragraoh:

"See section 9.1.4.3.2 for the Polar Crane seismic considerations. See section 3.8.3.4.10 for the applicable industry standard."

Insert the following paragraph following the 2nd paragraph:

"A telescopic jib crane is also provided inside Containment for refueling and maintenance. It is supported by a structural steel support mounted on the east-west divider wall between S. G. compartments at elevation 905'-9". Both the Polar Cane and Telescopic Jib Crane operating at the same time constitutes Multi-crane Operation which requires special administrative controls to reduce the likelihood of unplanned crane interactions. See section 3.8.3.1.2 for a description of the administrative controls and design features for the Polar Crane and the Telescopic Jib Crane."

FSAR Amendment 103b

LDCR-SA-2010-7, EV-CR-2010-003216-6 (SCD):

Modify the FSAR to provide the flexibility of using the normal Gas Decay Tanks in a Shutdown Gas Decay Tank capacity in the event that any of the designated Shutdown Gas Decay Tanks become degraded. The designated Shutdown Gas Decay Tanks are normally used during plant startup and shutdown. This system performs no function related to the safe shutdown of the plant. [11.3.2.1.1] This change has no safety impact on the plant.

LDCR-SA-2010-13, EV-CR-2010-002062-19 (TJD):

Add the following sentence to the end of FSAR section 2.4.13.1.5:

A Groundwater Monitoring Program is established and incorporates several sentinel wells in various locations throughout the plant site for prompt identification of potential radiological release source locations.

Justifcation - Update FSAR to include information on groundwater monitoring program from May 2010 hydrogeology study performed by Pastor, Behling, & Wheeler.

Change FSAR section 2.4.13.4 to read as follows (delete end of existing sentence #1, and add new sentence after existing sentence #1):

No planned releases to the ground water environment will take place at the plantsite. However a Groundwater Monitoring Program exists for early detection of inadvertent releases of radioactive material. Pertinent information is provided in Section 6.1 of the Environmental Report.

Justifcation - Update FSAR to include information on groundwater monitoring program from May 2010 hydrogeology study performed by Pastor, Behling, & Wheeler.

LDCR-SA-2010-9, EV-CR-2010-000988-14 (TJD):

Change Executive Vice President & Chief Nuclear Officer title to Senior Vice President & Chief Nuclear Officer.

Justification - Update of VP title.

Also, revise section 17.2.1.3.2, item 3., to read as follows:

Audit program review reports encompassed by Section 17.2.1.3.1(7) shall be forwarded to the Senior Vice President & Chief Nuclear Officer within 30 days following completion of the review.

Justification - Clarify the distribution requirements for audit program review reports.

LDCR-SA-2010-9, EV-CR-2010-000988-14 (TJD) (continued):

Section 17.2.18:

Change the second paragraph in section 17.2.18 to add the following sentence after the existing first sentence: "Audit reports shall be forwarded to the Senior Vice President & Chief Nuclear Officer and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization."

The new second paragraph will read as follows:

Planned and periodic audits are performed in accordance with written procedures to verify compliance with all aspects of the quality assurance program. Audit reports shall be forwarded to the Senior Vice President & Chief Nuclear Officer and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization. Responsibility for the evaluation program has been assigned to the Manager, Quality Assurance. Audits are conducted or coordinated by Quality Assurance personnel and shall include evaluation and examination of the following quality-related activities:

Justification - Clarify the distribution requirements for audit reports.

LDCR-SA-2011-1, EV-CR-2010-003005-5 (TJEW):

The FSAR will be revised to reflect the disconnection and abandonment of pressurizer heater elements #19, #20, and #45 and the corresponding reduction in heating capacity from 1800 kW to 1731 kW. This activity has no impact on the pressurizer design function as the minimum heater requirements are not affected.

On Table 5.1-1B (Unit 2) Sheet 1, revise the Pressurizer heater capacity, (kW) from 1,800 to 1,731.

FSAR Amendment 104

LDCR-SA-2010-18, EV-CR-2010-006120-13 (RAS):

Added the following sentence to the end of item #5:

"To prevent any adverse impact on controls at the HSP or local control stations when Control Room is evacuated due to a fire, the surveillance test program validates the opening of isolation contacts for transfer switches used to transfer controls in the event of a control room fire."

Clarifies surveillance testing requirements for controls used for shutdown from outside the Control Room.

Added note "(d)" to HS-4524FL and CS-BT-1EB13L

Added note "(d)" to HS-2452C

Added note "(b)" to HS-2452B

LDCR-SA-2009-14, EV-CR-2008-001609-00-11 (SCD):

Revise section 6.3.2.2.9.2 of the FSAR to include description of the new replacement SI to CL check valves (1/2SI-8819D). The replacement check valves are a nozzle style check as opposed to the currently described spring loaded, lift piston types for sizes 2" and smaller in the Emergency Core Cooling System.

LDCR-SA-2011-8, EV-CR-2010-011041-2 (SCD):

Insert the following prior to the last sentence in the second paragraph in section 11.4.2.7:

"Interim storage locations include a warehouse area (2K7) and a fenced area (3L15) shown in Figure 1.2-1."

The existing FSAR description does not reflect locations of interim radioactive material/ waste storage areas at CPNPP.

Revise Figure 1.2-1 to add a label for the Interim Radioactive Waster Storage Area 3L15.

LDCR-SA-2010-20, EV-CR-2010-008660-2 (TJD):

Correct description of PORV Tubing (between piping and PORV) and note that it applies to Unit 1 only. This change is required by regulation to match the as-built plant as shown on FSAR Figure 6.3-1 (M1-0262 sh. – and M2-0263-B). In this case, the Nitrogen Accumulators are inside a Code N-5 Boundary (i.e., are Code Class 3). Note 41 is only applicable to the tubing for Unit 1.

FSAR Amendment 104

LDCR-SA-2010-20, EV-CR-2010-008660-2 (TJD (continued):

Correct the line item for CCW air accumulators (as shown on FSAR Figure 9.3-3) to be consistent with all the other air accumulators and associated check valves and tubing as shown on FSAR Figure 9.3-1). As shown on the flow diagram, these accumulators are installed in the instrument air tubing system. This tubing was designed and installed in accordance with CPES-I-1018 and Table 17A-1 Note 41. Because they were installed in a tubing system, the check valves and accumulators were not included in an N-5 Code boundary. In accordance with T17A-1, Note 40, the Code Class must be "-".

Revise Reference FSAR Sections for Air Accumulators from Section 10.3 to Section 9.3.1 for consistency.

Revise Reference FSAR Sections for Tubing and supports from Section 9.3.1 to Section 3.9B for consistency.

Add Reference to FSAR Section 9.3.1 for Air Accumulators for consistency. Revise Reference FSAR Section for Tubing and supports from Section 9.3.1 to Section 3.9B for consistency.

Fix typographical error in Note 40. - "_" should be "-".

LDCR-SA-2008-21, EV-CR-2008-001221-00-12 (JCH):

Description of Change: Update FSAR to reflect radial arm hoist assembly (RAHA) QA Requirements.

Table 17A-1 Sheet 35 of 58: Changed Stud tensioner Handling Device to Radial Arm Hoist Assembly.

Table 17A-1 Sheet 58 of 58: Added Not 93.

Technical Justification:

FDA-2008-001221-04 is changing Unit 2 RAHAs to have quick disconnects for the hoists, similar to Unit 1. It also documents the current plan to remove the radial arms in Modes 1 - 4. The FSAR update is required to update the Q List for the pre-OL change from a Stud Tensioner Handling Device, which was a simple hoist, to the Radial Arm Hoist Assembly. The change is administrative and is required to document the applicable quality assurance requirements. Upgrade of the RAHA design to seismic Category II had been found impractical.

LDCR-SA-2010-5, EV-CR-2006-003080-00-320 (JDS):

Include reference in the discussion of Regulatory Guide 1.126 to reflect the vendor model for Westinghouse methods for the analysis of fuel densification used after the transition to the Westinghouse methodologies.

FSAR Amendment 104

LDCR-SA-2010-5, EV-CR-2006-003080-00-320 (JDS) (continued):

Add reference 12, the Westinghouse method (WCAP-8219-A) for the analysis of fuel densification used after the transition to the Westinghouse methodologies.

Remove reference 18 to Regulatory Guide 1.126, the Westinghouse method (WCAP-8219-A) for the analysis of fuel densification is used after the transition to the Westinghouse methodologies.

LDCR-SA-2010-8, EV-CR-2007-003164-00-3 (JDS):

Update title from "Fuel Handling Maching" to "Fuel Handling Bridge Crane" for consistency with other sections of the FSAR.

The changes are two electric trolleys hoist (one on East side and other one on West side) on an overhead structure instead of one electric trolley hoist. The hoist speeds are limited to 3 feet per minute in the slow zone and 21 feet per minute in the fast zone. For text of the change see attachment below. These changes reflect the current configuration of the new replacement Fuel Handling Bridge Crane.

In addition, the description of the features of this new crane (including speeds and automatic features) are included in the text.

This change identifies the code required ASME NUM-1 (Type B) for hoisting equipment and ASME NOG-1 (Type II) for trolley and crane structure for the design and construction of the new fuel handling bridge crane. For text of change see attachment below. The codes invoked encompass the requirements of CMAA-70 and include requirements typically recognized within the Nuclear Industry. Further, ASME NUM-1 and ASME NOG-1 codes apply to cranes operating in Nuclear Power Plant.

Item 33 of this Table:

Delete non-IEEE-383 cable and its PVC insulation associated with the fuel handling bridge crane anticrabbing control and computer control cables to the fuel handling bridge crane anticrabbing control TCS device. The anticrabbing control for existing crane will be removed along with the crane, since these will not be required for the new replacement crane.

LDCR-SA-2011-5, EV-CR-2009-000859-00-15 (JDS):

Changes to Quality Assurance section of FSAR to support implemtation of the Dry Cask Storage System to include the Independent Spent Fuel Storage Installation (ISFSI) facility, activities supporting spent fuel dry cask loading, and the associated dry cask storage equipment.

FSAR Amendment 104

LDCR-SA-2010-16, EV-CR-2010-008336-3(JCH):

Section 9.5.2 Communication Systems

Description: Update the FSAR to replace Private Automatic Branch Exchange(PABX) Telephone with Intraplant(IP) Telephone System in Section 9.5.2.

Page 9.5-71 Section 9.5.2.1 Change private automatic branch exchange (PABX) to Intraplant (IP).

Page 9.5-72 Section 9.5.2.2.2 Change PABX to IP.

Page 9.5-73 Section 9.5.2.2.5 Add IP and delete (PABX telephone system).

Page 9.5-74 Sections 9.5.2.2.7 and 9.5.2.2.10 Change PABX to IP.

Page 9.5-75 Section 9.5.2.2.10 Change PABX to IP.

Justification:

Administrative change by standardizing the name of Intraplant (IP) Telephone System used within FSAR which creates less confusion when telephone technology changes.

LDCR-SA-2011-3, EV-CR-2010-010560-2(SCD):

Update CPNPP management resumes and position descriptions.

LDCR-SA-2010-17, EV-CR-2009-002851-00-4(TJEW):

This activity does not provide a relaxation in the technical or maintenance requirements used when a Mild or Harsh equipment designation is applied. The FSAR is being revised to ensure that components installed within potentially harsh environments are consistently identified as harsh. It has no adverse impact and does not affect the safe shutdown capability of the plant, nuclear safety, industrial safety, plant, system or component reliability. No new test or experiments not described in the LBDs are associated with this activity.

Two CDF were identified as needing revision (22835 and 25701)

On page 3.11B-2,

1.) delete the second paragraph of the note under section 3 that reads,

"If radiation is the only harsh environment criterion exceeded for an area and an evaluation concludes that the component materials for a piece of equipment has a radiation threshold level greater than the postulated radiation environment, then, for purposes of environmental qualification, the equipment will be considered to be located in

LDCR-SA-2010-17, EV-CR-2009-002851-00-4(TJEW) (continued):

a mild environment. The radiation threshold level is the lowest radiation exposure at which property changes in component material is documented. Such documentation can be from materials handbooks, textbooks, government reports, laboratory data, and industry sources."

and

2.)delete the note in section 4 that reads,

"If relative humidity is the only harsh environment parameter exceeded for an area and an evaluation concludes that the subject equipment can perform its safety-related function(s) when exposed to the postulated relative humidity environment, then, for purposes of environmental qualification, the equipment will be considered to be located in a mild environment."

LDCR-SA-2009-4, EV-CR-2010-006971-17 (TJEW):

The installation of transfer panels will facilitate the connection of XST2A or XST2 to the 1E busses within the 72 hour LCO.

Revise transformer name from XST1/2 to XST2A in Figure 1.2-1.

In 8.2-10, revise transformer name from XST1/2 to XST2A on page 8-iv.

Revise transformer name from XST1/2 to XST2A on page 8.2-1.

Replact the paragraph on page 8.2-2 that reads,

"A spare startup transformer, XST1/2 with dual primary windings (345-kV and 138-kV), is stored in a dedicated location under the 345-kV line to XST2 (refer to Figure 8.2-1). This transformer must be physically relocated to replace XST2 or XST1 if required. This transformer is provided to prevent an extended interruption of offsite power in case of failure of any startup transformer."

with a paragraph that will read,

Spare startup transformer XST2A, with dual primary windings (345-kV and 138-kV), is in a dedicated location under the 345 kV line to XST2 (refer to Figure 8.2-1) to serve as a replacement of XST2. Cable buses from secondary X and Y windings of XST2 and XST2A are connected to two 6.9 kV transfer panels to provide 345 kV offsite power to Units 1 or 2 safety related buses. These transfer panels allow transfer of the 345 kV offsite power source for safety related buses from XST2 to XST2A and vice verse. This

LDCR-SA-2009-4, EV-CR-2010-006971-17 (TJEW) (continued):

spare transformer may be physically relocated to a dedicated location near XST1, to serve as a replacement of XST1.

In Figure 8.2-1,

1. revise the 345 transformer storage location name from "XST1/2" to " XST2A"

2. revise the 138 transformer storage location name from "XST1/2" to " XST1A"

3. correct the orientation and delete the extra transformer figure that does not exist next to 2ST

In Figure 8.2-4,

1. revise transformer name from XST1/2 to XST2A

2. swap name tags between XST2A and 1ST.

3. show a connection of 1ST to the non-safeguards buses and shorten the line representing the non-safeguards busses

4. revise the connections of XST2 and XST2A to show connection to the Transfer Panels

5. show the connection from the Transfer Panels to the 6.9 kV buses

6. relocate the "6.9KV SAFEGUARD BUSES (SEE FIGURE 8.3-1)" below the Transfer Panels

In Figure 8.2-5, Sheet 1, revise transformer name from XST1/2 to XST2A.

In Figure 8.2-10, Sheet 1,

1.) revise transformer name from XST1/2 to XST2A for the motor operarted switch, on the spare transformer, and on the Figure title

2.) add the new cable trench and cable bus from XST2A to the turbine building wall,

3.) Deleted "Amendment 100".

In Figure 8.2-10, Sheet 2,

1.) revise transformer name from XST1/2 to XST2A for the motor operated switch, transformer, and Figure title,

2.) Show the new cable trench and cable bus to the turbine building wall, and

LDCR-SA-2009-4, EV-CR-2010-006971-17 (TJEW) (continued):

3.) Deleted "Amendment 100".

In Figure 8.2-11,

1, revise transformer name from XST1/2 to XST2A

2, show the cable trench to from XST2 and XST2A to the Transfer Panels and to the 6.9kV buses.

In Table 8.3-3 Sheet 2,

1.) In "Item 4", under the "Function" column, add the words "through CPX-EPTSST-02Y" after Unit 1 and add the words

"through CPX-EPTSST-02X" after Unit 2.

2.) In "Item 4A", under the "Fuction" column, replace the words that read "buses 1EA1 and 1EA2 of Unit 1" with the words "transfer panel CPX-EPTSST-02Y".

3.) Under the "Effects on System" column, add the following words at the end of "Item 4" and "Item 4A", "Can also realign to spare startup transformer XST2A (Item 4C).

On Table 8.3-3, Sheet 3,

1.) In the "Function" column replace the words that say, "XST2 to buses 2EA1 and 2EA2" with the words "transfer panel CPX-EPTSST-02X from XST2",

2.) Add the words, "Can also realign to spare startup transformer XST2A (Item 4C)"

3.) Add the Failure Mode and Effect Analysis "4C" for the "Spare Startup transformer XST2A", "4D" for the "15-KV Cable" Y-winding connection, and "4E" for the X-winding connection.

In 9.5.1.5.6,

1. In the sentence that reads,

"The unit auxiliary transformer and the Unit 1 startup transformer are separated from the Turbine Buildings by a three-hr rated fire wall."

Delete the words "Unit 1" and after word "transformer" insert the words, "s XST1 and XST2"

2. Delete the sentence that reads,

LDCR-SA-2009-4, EV-CR-2010-006971-17 (TJEW) (continued):

"The Unit 2 startup transformer is also separated from the Turbine Building by a three-hr rated fire wall."

3. In the last sentence that reads,

" The Unit 1 station service transformer is separated from the Turbine Building by a distance greater than 50 feet."

Delete the words "The Unit 1", capitalize the "s" in station, and insert the following words after "transformer"

"s 1ST and 2ST and spare transformer XST2A are" and

delete the word "is"

On Figure 10.2-1 add "\XST2A" after XST2 in three locations.

LDCR-SA-2011-6, EV-CR-2009-000859-00-16 (JDS):

Include a description of the Independent Spent Fuel Storage Installation for the facility desciption for completeness.

Revise drawing to show location of the Independent Spent Fuel Storage Installation.

Update Table of Contents to reflect new section for Spenf Fuel Dry Cask Storage and clarify that section 9.1.2 is Spent fuel storage in the spent fuel pools and containment.

Update title to include description of spent fuel storage in "the spent fuel pools and containments."

Update text to refer to heavy load of the Holtec Hi-Trac tansfer cask for the Fuel building overhead crane.

Insert new section 9.1.5 to describe the independent spent fuel storage installation and the specific Certificate of Compliance (1014) used for dry cask storage.

Add an additional three references for the Cask FSAR, the Holtec Certificate of Compliance (1014) and the NUREG-6407 for Transportation and packaging of spent fuel.

Add reference to the Fire Hazards Analysis for ISFSI and Haul Path to the ISFSI.

Include reference to the CPNPP 10CFR72.212 Evaluation report.

Include statement for a fully loaded (84 Casks) bounding annual exposure for the ISFSI in the estimated annual dose at the exclusion boundary and to the population at large.

LDCR-SA-2011-12, EV-CR-2010-008829-2 (TJEW):

On Figure 9.5-52 (M1-0215), Sheet F:

1.) add note 10 that says, "10. Alternate fill location connection point. Box with Access Cover Plate is maintained watertight"

2.) On the truck fill watertight valve box add "COVER" in between "REMOVABLE" AND "PLATE"

3.) add an arrow with a note that says "See Note 10" at the quick disconnect fitting

4.) at the quick disconnect fitting add "COVER" in between "REMOVABLE" AND "PLATE"

5.) add the removable access cover plate and add an arrow pointing to the removable access cover plate and the note that says, "REMOVABLE ACCESS COVER PLATE"

On Figure 9.5-52 (M2-0215), Sheet F:

1.) add note 12 that says, "12. Alternate fill location connection point. Box with Access Cover Plate is maintained watertight"

2.) On the truck fill watertight valve box add "COVER" in between "REMOVABLE" AND "PLATE"

3.) add an arrow with a note that says "See Note 12" at the quick disconnect fitting

4.) at the quick disconnect fitting add "COVER" in between "REMOVABLE" AND "PLATE"

5.) add the removable access cover plate and add an arrow pointing to the removable access cover plate and the note that says, "REMOVABLE ACCESS COVER PLATE"

On Figure 9.5-52 (M1-0215), Sheet G:

1.) add note 10 that says, "10. Alternate fill location connection point. Box with Access Cover Plate is maintained watertight"

2.) On the truck fill watertight valve box add "COVER" in between "REMOVABLE" AND "PLATE"

3.) add an arrow with a note that says "See Note 10" at the quick disconnect fitting

4.) at the quick disconnect fitting add "COVER" in between "REMOVABLE" AND "PLATE"

5.) add the removable access cover plate and add an arrow pointing to the removable access cover plate and the note that says, "REMOVABLE ACCESS COVER PLATE"

LDCR-SA-2011-12, EV-CR-2010-008829-2 (TJEW) (continued):

On Figure 9.5-52 (M2-0215), Sheet G:

1.) add note 12 that says, "12. Alternate fill location connection point. Box with Access Cover Plate is maintained watertight"

2.) On the truck fill watertight valve box add "COVER" in between "REMOVABLE" AND "PLATE"

3.) add an arrow with a note that says "See Note 12" at the quick disconnect fitting

4.) at the quick disconnect fitting add "COVER" in between "REMOVABLE" AND "PLATE"

5.) add the removable access cover plate and add an arrow pointing to the removable access cover plate and the note that says, "REMOVABLE ACCESS COVER PLATE"

On page 9.5-81, Section 9.5.4.2.1 add the following words to:

1.) at the end of the first sentence of the forth paragraph add, "or via an alternate fill method which utilizes a special diffuser tool to fill through the Sample Grab Point quick-disconnect fitting connection, which is also in a watertight box." and

2.) After the third sentence of the fourth paragraph add the sentence, " When the alternate fill method is used, the special diffuser tool is also capped at the bottom and perforated with holes."

On figure 9.5-51,

1.) add note 1 that says, "Note 1. Sample Grab Point Connection provides additional truck fill method via use of special diffuser tool. This box is maintained watertight." in the lower left hand corner and

2.) add an arrow pointing at the Sample Gab Point Connection and a text box that says, "See Note 1"

LDCR-SA-2010-19, EV-CR-2010-010870-1 (TJEW):

On page 10.4 page 33, third paragraph, replace the word "each" with "Unit 2" and add a "s" after the first occurrence of the word "generator".

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW):

Justification: Changes to the description of the 345 kV Switchyard are required for addition of two breakers and their associated lines (Everman & Parker 2) and relocation of the TO's breaker controls/protection to TO's control building. With the new relay control panels a third set of protective relays are not required for line breakers. Therefore, the lines will only have primary or secondary (backup) sets. In addition, the

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW) (continued):

FSAR and associated Figures are being revised to reflect name changes of TXU Electric Delivery to Transmission Operator (TO) in the affected sections. Further, the approximated length of transmission lines to the substations connected to the CPNPP Switchyards is deleted, because this detail is not required for the FSAR. Also the discussion of transmission towers on pages 8.2-3 and 8.3-4 is being deleted from the FSAR because the information is shown in Figure 8.2-12. Section 8.2.2 analysis is updated to reflect the impact of loss of double circuit towers as a result of addition of new lines. In addition Oncor has reviewed the the section 8.2.2 analysis for its adequacy due to the addition of new lines.

On Page 1.2-9, second paragraph, section 1.2.2.5

1.) change "five" to "seven" and

2.) In the same paragraph, change "owner's" to "Transmission Operator's"

On Page 8.1-1,

1.) Update the Utility Grid Description to reflect the changes in the ERCOT organization.

2.) change all of "TXU Electric Delivery" to "TO"

On Page 8.2-1, change

1.) all of "TXU Electric Delivery" to "TO" and

2.) the reference to "five" 345kV transmission lines to "seven" lines and

3.) In the third paragraph, add the word system before the words "as shown on Figure 8.2-4"

4.) Add the following sentence, from page 8.2-3, to the end of the third paragraph, "The 138-kV switchyard is physically and electrically independent of the 345-kV switchyard."

On Page 8.2-2,

1.) In the 4th paragraph, delete the words that say, "The network transmission line terminals of 345-kV switchyard are provided with a third set of relay protection to enhance the reliability of transmission system fault isolation."

2.) In the 5th paragraph, delete the words that say, "A separate 125-VDC system is provided for the third set of relay protection in the 345-kV switchyard."

3.) In the 6th paragraph,

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW) (continued):

a.) add the following words, "The 345-
kV switchyard TO circuit breakers may be operated from either the 345-
kV Switchyard Control Building or remotely from TO's System Operations Center."

b.) In the next sentence change the word "all" to "CPNPP", delete the words "the spare", and add the word XST2A after transformer.

c.) In the next sentence capitalize the first letters in the following words, "switchyard relay house", add "138-kV" before the words

"switchyard relay house", and replace the words "through TXU Electric Delivery" with "from TO".

4.) In the 8th paragraph,

a.) delete the words, "and the approximate length of their respective transmission lines", rearrange the list of tranmission lines to match , as shown below, the switchyard Figure 8.2-4, and delete the transmissison line lengths.

"* DeCordova (138-kV)

* Stephenville (138-kV)

* DeCordova (345-kV)

* Wolf Hollow (345-kV)

* Everman (345-kV)

* Johnson Switch (345-kV)

* Comanche Switch (345-kV)

* Parker No. 1 (345-kV)

* Parker No. 2 (345-kV)"

On page 8.2-3,

1.) The order of the CPNPP transmission lines, including the Stephenville line, will be revised as shown on Page 8.2-2 and therefore, the Stephenville line detail on the first line of this page will be deleted.

2.) Insert the following words before the first paragraph,

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW)(continued):

"The layout of transmission lines from TO’s CPNPP switchyards to other switching stations, in the vicinity of CPNPP switchyards, is shown in Figure 8.2-12.

The following combinations of lines form double-circuits routed on common transmission towers:

345-kV Parker No. 1 and 345-kV Parker No. 2

345-kV Comanche Switch and 138-kV Stephenville

345-kV Everman and 345-kV Johnson Switch

345-kV DeCordova and 345-kV Wolf Hollow

3.) Delete the next 6 paragraphs.

4.) In the 7th paragraph,

a.) delete, the 1st sentence,

b.) in the 2nd paragraph, add "138-kV" after "The".

c.) delete the third sentence.

5.) In the 8th paragraph,

a.) Delete the first sentence,

b.) in the second sentence, replace "The line then" with "The 138-kV line to Stephenville", add "Everman," after "Wolf Hollow", and replace "Venus before making another 90° turn" with "Johnson Switch".

c.) Delete the 3rd sentence.

On page 8.2-4, delete the words from the paragraph that started on page 8.2-3, "Upon making the second turn, the line then shares common towers with the 345-kV line to Comanche Switch."

On Page 8.2-5, change all of "TXU Electric Delivery" and "TDSP" to "TO".

On page 8.2-7,

1.) second paragraph of section 8.2.2,

a.) add a comma in the first sentence after the words "Comanche Switch line"

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW) (continued):

b.) in the two places where "Parker Line" is written, before the word "line" insert, "No. 1 and Parker No. 2 double-circuit", in the three places where the word "Venus" is written, replace it with "Everman and Johnson Switch double-circuit", add the word "the" before the second occurrence of "Venus"

c.) In the 4th sentence, add a "-" between "double" and "circuit"

2.) In the third paragraph of section 8.2.2

a.) delete the words "Electric Reliability Council of Texas" and the "()" around ERCOT

b.)change "TXU Electric Delivery" to "TO"

On Page 8.2-8,

1.) change "TXU Electric Delivery" and "TXU Delivery" to "TO".

2.) Delete the entire 7th paragraph.

On Page 8.2-9,

a.) change all of "TXU Electric Delivery" to "TO".

B.) Replace the words, "There are no restrictive operating limits (real and reactive power, voltage, frequency and other) established for CPSES by ERCOT. However, CPSES requires that TXU Electric Delivery transmission" with "In order to satisfy offsite power requirements, the TO should"

On Page 8.3-2, second paragraph, change

1.) "TXU" to "TO" and

2.) the reference to "five" 345kV transmission lines to "seven" lines

Revise Figure 8.2-1 to show the new 345 kV Parker and Everman Lines, delineate the Parker No. 1 and No. 2 lines, add the TO Control Building, and change the Venus line to Johnson Switch.

Revise Figure 8.2-4 to show the new 345 kV Parker and Everman Lines and Breakers at both ends of the lines, delineate the Parker No. 1 and No. 2 lines, change TXU Energy Delivery to TO, change the Venus line to Johnson Switch, and revise the breaker arrangement at Johnson Switch and Wolf Hollow.

Revise Figure 8.2-5, sheet 1 to show

1.) to show the new Oncor Control Building

LDCR-SA-2009-20, EV-CR-2009-005301-7 (TJEW) (continued):

2.) the new West Bus, Bay 9 Breaker for the Parker No. 2 transmission line

Revise Figure 8.2-5, sheet 2 to show

1.) change "Venus" to "Johnson Switch"

2.) the addition the new 345 kV Everman Line Breaker and associated equipment on the East Bus of bay 4, change the Venus line to Johnson Switch

Revise Figure 8.2-12 to show

1.) the new 345 kV Parker and Everman Lines,

2.) delineate the Parker No. 1 and No. 2 lines,

3.) change the Venus line to Johnson Switch.

LDCR-SA-2010-1, EV-CR-2006-003080-44 (SCD):

The changes made to FSAR Section 11 reflect the change in isotopic inventory associated with the increase in power as well as the transition from historical TXU calculation methodologies to those of Westinghouse.

The changes made to FSAR section 12 reflect the revised dose estimations associated with normal operations and assuming the revised isotopic inventory. Note dose assessments associated with accident conditions, i.e., FSAR Chapter 15 events, were performed by Westinghouse and are outside of the scope of this LDCR.

FSAR Amendment 104a

LDCR-SA-2010-14, EV-CR-2007-003115-00-12 (GLM):

The FSAR is being revised because the existing clarifier based pre-treatment subsystem portion of the Water Treatment (WT) System has deteriorated due to the aggressive nature of the lake water and chemicals required for treatment. The pre-treatment subsystem portion of the WT System is being replaced using current microfiltration technology.

With this modification, the clarifier is being taken out of service and replaced with microfiltration units. Therefore, changes to the system description are required, which maintains the design requirements. This modification removes the clarifier, pressure filters, and filtered water forwarding pumps from the WT System and replaces it with microfiltration units. These units are a membrane based filtration system which removes suspended solids and achieves water quality requirements in a one-step filtration process. The system includes filtration tanks, pump skid, module skid, and a clean-in-place (CIP) skid. These changes reflect the new equipment being installed and maintain the design requirements.

LDCR-SA-2011-13, EV-CR-2010-000979-15 (JDS):

Delete TM symbol from Zirlo as this a Registered tradmark. Zirlo is now used to refer to this type of cladding material.

Add rod bow penalty WCAPs (8691 and 8692) to the material incorporated bny reference. Text added to chapter 4 describes this as the accepatable methodology for use to calculate rod bow penalty.

Include the discussion of the use of the Standardized Debris Filter Bottom Nozzle (SDFBN) and the Robust Protective Grid (RPG). These fuel assembly components are now characteristic for reloads at CPNPP.

Update Minimum DNBR for design transients to reflect appropriate values based on WRB correlation used. Update the Average mass velocity for correct description for the values provided. Update maximum overpower value used for Peak linear power resulting from overpower transients/operator errors. Insert RPG for robust p grid usage.

Include the discussion of the use of the Robust Protective Grid (RPG). This fuel assembly components is now characteristic for reloads at CPNPP.

Provide a description of the Standardized Debris Filter Bottom Nozzle (SDFBN) for Westinghouse 17x17 fuel. This design is now standard for reloads for CPNPP units 1 and 2.

Provide a description of the Westinghouse Robust Protective Grid (RPG) design. This design is now standard for reloads for CPNPP units 1 and 2.

Clarify use of nomenclature used for Westinghouse fuel for the Vantage+ fuel design.

FSAR Amendment 104a

LDCR-SA-2011-13, EV-CR-2010-000979-15 (JDS) (continued):

Update discussion on Effects of Rod Bow on DNBR to reflect NRC approved methodology.

Include reference to Effects of Rod Bow on DNBR approval by the NRC (WCAP 8691 and 8692)

Update maximum overpower value used for Peak linear power resulting from overpower transients/operator errors.

LDCR-SA-2011-14, EV-CR-2011-001279-1 (JDS):

Update 7 minutes 30 seconds to 8 minutes and 13 minutes to 14 minutes for consistency with previous statement on this same page and the supporting analysis.

LDCR-SA-2011-2, EV-CR-2009-001766-00-4 (RAS):

Add the following to the end of the sentence for item 'b.' under "Testing of Reactor Trip Breakers":

"...or by a Reactor Trip Test pushbutton via a keylocked test selector switch."

LDCR-SA-2009-22, EV-CR-2007-000293-00-1 (TJD):

FSAR section 17.2.15 is being changed to utilize a conditional release to not only install the nonconforming component in the plant but also to declare the affected SSC Operable per Technical Specifications (TS), provided that resolution of the nonconforming condition is tracked in the Corrective Action Program (CAP), and an evaluation of the nonconforming condition shows that it does not adversely affect Operability of the component. This change was approved by the NRC in a Safety Evaluation dated 9/23/11.

Revise item 5 as follows:

Conditional releases allow issuance of nonconforming items from the warehouse for installation, testing, and operation, pending disposition of the nonconformance. For nonconforming items installed with a conditional release, and affecting Technical Specifications, credit may be taken for Technical Specification operability of the item, provided that: a) resolution of the nonconforming condition is tracked, and b) evaluation of the nonconforming condition supports operability of the component. Each conditional release also describes any limitations or special precautions required. Conditional releases are periodically evaluated as to their status and the results forwarded to management for their review.

FSAR Amendment 104a

LDCR-SA-2011-15, EV-CR-2008-003340-00-6 (TJEW):

Table 3A.3-2, Sheet 5, section 1.3, is changed via FDA-2008-003340 and in the FSAR via LDCR SA-2011-015 in EV-CR-2008-003340-6 to revise boron concentration from '2200' to '2600' and from "initial" to "final" spray solution to (1) align configuration control and (2) correct the statement so that it is understood that short period exposure to out of specification ph chemical spray will have no effect on the ability of equipment inside containment to perform their safety function.

LDCR-SA-2009-18, EV-CR-2009-001588-00-31 (TJEW):

Technical Justification:

Issue 1 - FSAR Table 7.5-7C note 21 and the response to the TMI Action Plan Section II.F.1 state that the "containment water level covers the entire range of expected water level in the containment for post-accident conditions." This statement regarding the wide range span and the requirement for a separate narrow range instrument was based on a misinterpretation of the maximum flood level calculation 525-00. The calc had conservatively assumed that the reactor cavity was not flooded to maximize the peak flood level. Based on Calculation 525 dated 3-23-1979, the range of containment flooding was calculated from El. 808 to El. 817'-6". Based on the current calculation NU(B)-057 R3, the range of flooding is from the El. 781'-2" rector cavity sump and El 783'-7" reactor cavity [see A1-0529] to the maximum flood El. 816'-10". The FSAR requires update and clarification.

Issue 2 - FSAR table 7.5-7E note 18 states "Containment Water level transmitters are multipoint sensors and failure of any one sensor would not cause ambiguity and hence will not have any adverse effect on the monitoring of containment water level." This statement was based on the assumption that the analog control board indicators for the original level instrumentation would be multipoint LEDs corresponding to the RTDs. However, the original 4-20 ma analog indicators were not replaced. The failure mode of the loop is such that the failure of one sensor can be read as one foot high or one foot low across the scale. This is an ambiguous failure. The FSAR requires update and clarification.

On page 7.5-10, in Section 7.5.2.1, replace "10. Containment Water Level" with the word "Deleted"

In Table 7.5-2, delete "Containment Water Level Key A1"

In Table 7.5-7A, Sheet 2 of 15, for the row containing "CONTAINMENT WATER LEVEL (NOTE 21), delete "A1" and add "-3"" after "808'".

In Table 7.5-7C, Sheet 2 of 4, number 21, 1.) Insert the words "(from prior to the start of switchover from injection to recirculation to the maximum containment flood level)" between the words "level" and "in" and 2.) After the first sentence, insert the sentence, "The water level in the sump is assured by the RWST Low-Low level setpoint for the start of switchover. Operator action based on containment water level is not required."

LDCR-SA-2009-18, EV-CR-2009-001588-00-31 (TJEW) (continued):

In Table 7.5-7D, Sheet 1 of 7, and in the row of "Containment Water Level" add "-3"" after "808'".

In Table 7.5-7E, Sheet 3 of 3, number (18), delete all the words after "transmitters" and add the following words, "were intended to be multipoint sensors which would prevent an ambiguous failure from occurring. The original indicators were never replaced, however, and the original analog indicators rely on an additive logic to compute the water level. This results in the potential for a failed indicator to cause a false reading of ± 1 foot. The containment water level indicators are for information only and will not adversely affect the ability to safely mitigate an accident."

In section 11.F.1, CPNPP Response (5) on page II.F-3, 1.) Add "-3"" after "808'" 2.) Add the words "the intent of" between "meeting" and "Regulatory" 3.) replace the words "entire range of expected water level in the Containment for post accident conditions" with the words, "maximum expected flooding levels assuming the reactor cavity is blocked and does not receive any flood water to simulate the most limiting post-accident condition (flooding starts at El. 808'). Actual post-accident conditions would include flooding into the reactor cavity sump (781' -2") and reactor cavity (783' - 7").

LDCR-SA-2011-18, EV-CR-2011-008509-4 (TJEW):

Revise FSAR Section 9.5.5.2 to reflect information in FSAR prior to issuance of LDCR-SA-2004-014 which changed the Jacket Water Heat Exchanger leakage rate. The justification for this change is that during the 2011 PI&R Inspection, a potential Severity Level IV violation of 10CFR50.59 was identified for an incorrect 50.59 screen related to changing the EDG jacket water leakage rates in the FSAR. The screen incorrectly concluded that an evaluation was not required.

On page 87 of sectio 9.5.5.2, 6th paragraph, change 408 gallons to 310, change 17 gallons per hour to 1.5, and change 24 hours to seven (7) days.

LDCR-SA-2011-21, EV-CR-2011-014200-1 (TJEW):

Components FT-6708 and FT-6709 are identified within DBD-EE-004 and FSAR Table 7.5-7B (Sheet 8 of 11) as being required to be seismically qualified. The devices are in fact only required to be seismically mounted. DBD-EE-004 section 4.3.2.1.1 states 'Category 2 instrumentation supplied from Non-1E power need not be seismically qualified unless it is part of a seismic Category I system and its failure may degrade this system.' The flow transmitters are N1E, Process Instruments(Isolable, Offline). DBD-ME-028, attachment 19, page 7 of 23, identifies that these types of transmitters are ANSI class: N/A, IEEE Class: N1E, Seismic Category: None, Safety Related: No, and EQ: No.

DBD-ME-028, Table 1, note 3, is associated with components identified as N1E, Process Instruments(Isolable, Offline). This note states that Seismic Category II mounting applies to the non-1E transmitter or gauge. This mounting requirement is identified within Maximo-MEL. The Location screen contains attribute 'CFC2'. For FT-6708 and FT-6709

FSAR Amendment 104a

LDCR-SA-2011-21, EV-CR-2011-014200-1 (TJEW) (continued):

the components are identified as SMTQ. The SMTQ designation identifies that the components require 'Seismic Integrity Evaluation and Mounting of Non-safety Related Instruments Connected to ANSI 2 or 3 Fluid Systems'.

Revise the note in Table 7.5-7B, Sheet 8 of 11, Variable - Safety Chilled Water Flow from "Note 5" to "Note 6"

LDCR-SA-2011-17, EV-CR-2011-003925-5 (JDS):

Reference Westinghouse Letter transmitting the GOTHIC containment pressure and temperature analyis for EQ.

Remove curves ad replace with methods. Reflects that the containment temperature and pressure curves used for EQ are no longer the same as those used for the containment integrity analysis.

LDCR-SA-2011-7, EV-CR-2009-000859-00-34 (JDS):

Update Table 3.7B-2 with newer versions of the ANSYS program. These versions of ANSYS were used for ancillary equipment used in the fuel building supporting the Dry Cask Storage project.

LDCR-SA-2011-16, EV-CR-2011-005954-4 (SCD):

Incorporate change of Quality Assurance (QA) to Nuclear Oversight.

FSAR Amendment 104b

LDCR-SA-2012-9, EV-CR-2011-012481-1 (RAS):

Revise the paragraph at the top of the page as follows to reflect the implementation of the Relaxed Axial Offset Control Methodology:

The axial flux difference deviation alarms are derived from the plant process computer which determines the 1 minute averages of the excore detector outputs to monitor delta flux in the reactor core and alerts the operator where delta flux alarm conditions exist. When power level is 50 percent or greater, an alarm message is output immediately upon determining a delta flux exceeding the Technical Specifications (this alarm requires 2 of the 4 AFD channels to exceed the power dependent limits). The signals from the four section excore detectors are summed and calibrated to a power measurement made using a secondary system calorimetric. The signals from the upper two sections and lower two sections are combined to give an upper flux and lower flux signal as shown in Figure 7.1-2. These upper and lower signals must be calibrated to agree with axial offset measurements derived from flux maps made using the movable incore system. The NIS provides both slope and zero-offset adjustments to achieve this calibration.

LDCR-SA-2012-7, EV-CR-2012-002362-1 (TJEW):

Background: During preparation of FDA-2011-000004-01, an error concerning Flow Diagram M2-0263, Sheet No. C, Safety Injection System Sheet 6 of 6, was found. It is not included in FSAR Figure 6.3-1. This flow diagram is part of the Safety Injection System and Containment Isolation System described in the FSAR Chapter 6 text and tables. Without it, FSAR Figure 6.3-1 is incomplete.

Non-conforming Condition: The Note block designating the drawing as part of FSAR Figure 6.3-1 is missing. Therefore, the Drawing was not included in the FSAR figure as required by RG 1.70 R2 and the FSAR is not current as required by10CFR50.71(e). The FSAR is not being update for changes to the facility as required by regulations (e.g. FDA-2008-003459-04). Corrective Action required: M2-0263 Sheet No. C must be revised and added to FSAR Figure 6.3-1 and to the Chapter 6 TOC.

Additionally, while performing a consistency review of the LDCR, found that FSAR Tables 3.2-3 and 3.2-4 also need to include M2-0263-C.

On Table 3.2-3, Sheet 10 fo 13, there is no entry for Unit 1; however, for Unit 2 add, "M2-0263-C CP-7 104* 6.3-1"

On Table 3.2-4, Sheet 1 of 16, there will be no entry for Unit 1; however, for Unit 2 add "6.3-1 M2-0263-C"

On page 6-xvii, the Table of "List of Figures" add ", M2-0263-C" at the end of the title for 6.3-1.

FSAR Amendment 104b

LDCR-SA-2012-11, EV-CR-2011-001492-12 (JCH):

Description: Revise Table 3.9N-10, "Unit 1 & 2 Active Valves," to include active relief valves 1-8121 and 2CS-8000.

Justification: RCP seal water return line relief valves, 1-8121 and 2CS-8000, are required to protect the excess letdown/seal water return header from overpressurization in the event that the line exiting containment is isolated while the excess letdown and/or the RCP No. 1 seal leakoff continues in service. The IST plan correctly designates these valves as active for Containment Penetration Overpressure Protection from both seal injection and thermal expansion however DBD-ME-255 Table 4-1 and FSAR Table 3.9N-10 show these valves as passive. This change will align the design and licensing basis documents to the IST plan. There is no change to the valve or its subcomponents and no change to the design function of the valve.

LDCR-SA-2012-10, EV-CR-2012-000999-1 (SCD):

Description of the Old Steam Generator Storage Facility was not adequately incorporated into the proper FSAR chapters.

LDCR-SA-2012-14, EV-CR-2011-008520-14 (SCD):

The Safe Shutdown Impound (SSI) is a thriving dynamic biological system with significant biological populations that can contribute to debris entering the Station Service Water (SSW) system. This debris can result from die-off of small fish (i.e. Threadfin Shad or Silver Sides), algae, live or dead Harris mud crabs and storm water runoff debris. Shad die-offs have occurred in the SSI that caused challenges to the SSW system safety related strainers. The SSI communicates with Squaw Creek Reservoir (SCR), which allows exchange of fish species and complicates measures to reduce or control fish populations. The installed floating booms minimize the ability of floating debris to enter the SSW system while the equalization canal net minimizes the migration of fish from SCR into the SSI.

Clarification is required on the limitation of use of the stop gates as a sole source of flood protection of the Turbine Building when the Circulating Water System is opened for maintenance. Due to the access opening in the operating deck of the circ-water discharge structure, the stop gates only isolate the CW Tunnels and any open CW system when the lake level is below 778 feet. As a result, any open CW system at elevation below the PMF level of 789.7 feet must be reassembled/closed or isolated when the lake level is on a rise and evident that it will reach/exceed a level of 778 feet. This ensures the isolation of the lake levels above 778 from the open pathway that could lead to flooding of the Turbine Building and lower level of the Electrical & Control Building on Elevation 778.

FSAR Amendment 104b

LDCR-SA-2010-3, EV-CR-2010-000528-8 (TJEW):

On page 1A(B)-55, in note a at the bottom of the page, change "0.8348" to "0.8299" and "38" to "39"

In 1A(B) on page 56, Part 9.C.2.a, change "0.8348" to "0.8299" and "38" to "39"

On page 9.5-82, change "0.8348" to "0.8299" and "38" to "39"

On page 9.5-83, change "0.8348" to "0.8299" and "38" to "39"

On page 9.5-86, change "0.8348" to "0.8299" and "38" to "39"

LDCR-SA-2012-17, EV-CR-2011-008700-17 (SCD):

Clarification is required on the limitation of use of the stop gates as a sole source of flood protection of the Turbine Building when the Circulating Water System is opened for maintenance. Due to the access opening in the operating deck of the circ-water discharge structure, the stop gates only isolate the CW Tunnels and any open CW system when the lake level is below 778 feet. As a result, any open CW system at elevation below the PMF level of 789.7 feet must be reassembled/closed or isolated when the lake level is on a rise and evident that it will reach/exceed a level of 778 feet. This ensures the isolation of the lake levels above 778 from the open pathway that could lead to flooding of the Turbine Building and lower level of the Electrical & Control Building on Elevation 778.

LDCR-SA-2012-15, EV-CR-2010-011195-4 (SCD):

10CFR50.71(e) requires the FSAR to be maintained current. This is an administrative update to the FSAR.

The FSAR 1A(B) exceptions to Revision 1 of RG 1.52 were reconciled with the RG and FSAR Table 6.5-1.

This Appendix 1A(B) discussion of RG 1.197 was not correct. It stated the TSTF incorporates the specific aspects of the RG whereas the TSTF took exception to specific aspects of RG1.197. This error was made at some point after LA 136.

The following are CPNPP exceptions to Sections C and C.1.2 of RG 1.197 documented in TS 5.5.20 which are not reflected in FSAR Appendix 1A(B):

1. C. - Section 4.3.2 “Periodic CRH Assessment” from NEI 99-03 Revision 1 will be used as input to a site specific Self Assessment procedure.

LDCR-SA-2012-15, EV-CR-2010-011195-4 (SCD) (continued):

2. C.1.2 - No peer reviews are required to be performed.

This Appendix 1A(B) discussion of RG 1.197 was not correct. It stated the TSTF incorporates the specific aspects of the RG whereas the TSTF took exception to specific aspects of RG1.197. This error was made at some point after LA 136.

The following are CPNPP exceptions to Sections C and C.1.2 of RG 1.197 documented in TS 5.5.20 which are not reflected in FSAR Appendix 1A(B):

1. C. - Section 4.3.2 “Periodic CRH Assessment” from NEI 99-03 Revision 1 will be used as input to a site specific Self Assessment procedure.

2. C.1.2 - No peer reviews are required to be performed.

The FSAR 1A(B) exceptions to Revision 1 of RG 1.52 were reconciled with the RG and FSAR Table 6.5-1.

LDCR-SA-2011-11, EV-CR-2009-001966-11 (GLM):

In the first sentence of Section 9.4C.6, "OFFICE AND SERVICE AREA HVAC SYSTEM" delete the words "...and relative humidity." In the first sentence, second paragraph of Section 9.4C.6, "OFFICE AND SERVICE AREA HVAC SYSTEM" change "air conditioning units" to "air handling units." Delete the second sentence, second paragraph of Section 9.4C.6, "OFFICE AND SERVICE AREA HVAC SYSTEM." In the third sentence, second paragraph of Section 9.4C.6, "OFFICE AND SERVICE AREA HVAC SYSTEM" change " direct expansion refrigerant cooling coil" to "chilled water cooling coil" and delete the word "humidifier."

The replacement air handling units are rated to provide a level of cooling which exceeds the rated capacity of the previous split air-conditioners (109.4 tons versus 94.5 tons) therefore this change does not adversely impact any design function of the Office and Service Area system, and the resulting changes to the FSAR are required to maintain configuration control. The humidifier does not perform or support a safety-related function for the O&SA HVAC system, therefore this change is not adverse. These changes do not adversely impact the O&SA HVAC system design function which is to provide a comfortable environment for operating personnel by maintaining the temperature and humidity as indicated by UFSAR Table 9.4-2.

For the Office and Service Area A/C, change the Relative Humidity (%) from "35-50" to (b).

The replacement air handling units are rated to provide a level of cooling which exceeds the rated capacity of the previous split air-conditioners (109.4 tons versus 94.5 tons) therefore this change does not adversely impact any design function of the Office and Service Area system, and the resulting changes to the FSAR are required to maintain configuration control. The humidifier does not perform or support a safety-related function for the O&SA HVAC system, therefore this change is not adverse. These changes do not adversely impact the O&SA HVAC system design function which is to provide a comfortable environment for operating personnel by maintaining the temperature and humidity as indicated by UFSAR Table 9.4-2.

LDCR-SA-2011-19, EV-CR-2009-001930-5 (GLM):

Revise the last sentence to state that the potable water supply is from the Somervell Co. Water District public water system and groundwater wells are a backup.

Justification:

The station potable and sanitary water system is designed to provide the following: 1) Water for toilets, sinks, showers, and drinking purposes in all permanent personnel areas of the plant site, as required, 2) Water for emergency eyewash and showers, as required, 3) Water to fire protection hoses for various on-site buildings, 4) Water to fill and to provide normal makeup to the Fire Protection Storage Tanks. The change requested by this LDCR is a change of source which is authorized under SFER-4205839. The changes will not have adverse impact to the design bases described in FSAR Section 4.2.5.1.

LDCR-SA-2011-19, EV-CR-2009-001930-5 (GLM) (continued):

Using SCWD as the primary source, the plant potable water distribution piping will function as it currently does.

As discussed in FSAR Section 9.2.4.3, this system is not connected to any radiological systems. Since the system is physically independent of radiological systems, this change to the potable water system has no impact on the plant radiological systems. Should a radiological spill occur, the conclusions of the FSAR for the potable water system are not changed - i.e., the change to the distribution piping is in the same general vicinity as existing potable water distrubition piping and cannot be directly impacted from a radiological spill outside the plant. Therefore the requested change does not adversely affect any FSAR described design function.

Delete the first sentence and revise the second paragraph to reflect that potable water is chlorinated to conform to TDH standards and the emergency shower in the relay house is chlorinated.

Justification:

The station potable and sanitary water system is designed to provide the following: 1) Water for toilets, sinks, showers, and drinking purposes in all permanent personnel areas of the plant site, as required, 2) Water for emergency eyewash and showers, as required, 3) Water to fire protection hoses for various on-site buildings, 4) Water to fill and to provide normal makeup to the Fire Protection Storage Tanks. The change requested by this LDCR is a change of source which is authorized under SFER-4205839. The changes will not have adverse impact to the design bases described in FSAR Section 4.2.5.1. Using SCWD as the primary source, the plant potable water distribution piping will function as it currently does.

As discussed in FSAR Section 9.2.4.3, this system is not connected to any radiological systems. Since the system is physically independent of radiological systems, this change to the potable water system has no impact on the plant radiological systems. Should a radiological spill occur, the conclusions of the FSAR for the potable water system are not changed - i.e., the change to the distribution piping is in the same general vicinity as existing potable water distrubition piping and cannot be directly impacted from a radiological spill outside the plant. Therefore the requested change does not adversely affect any FSAR described design function.

LDCR-SA-2012-2, EV-CR-2012-000111-3 (TJEW):

Justification:

Components FT-142, FT-143, FT-144, and FT-145 are identified within DBD-EE-004 and FSAR Table 7.5-7B (Sheet 3 of 11) as being required to be seismically qualified. The devices are in fact only required to be seismically mounted. DBD-EE-004 section 4.3.2.1.1 states 'Category 2 instrumentation supplied from Non-1E power need not be

LDCR-SA-2012-2, EV-CR-2012-000111-3 (TJEW)(continued):

seismically qualified unless it is part of a seismic Category I system and its failure may degrade this system.' The flow transmitters are N1E, Process Instruments(Isolable,

Offline). DBD-ME-028, attachment 19, identifies that these types of transmitters are ANSI class: N/A, IEEE Class: N1E, Seismic Category: None, Safety Related: No, and EQ: No.

DBD-ME-028, Attachment 19, Table 1, note 3, is associated with components identified as N1E, Process Instruments(Isolable, Offline). This note states that Seismic Category II mounting applies to the non-1E transmitter or gauge. This mounting requirement is identified within Maximo-MEL. The Location-Qualification screen denotes a Component Function Code 2 ('CFC2') attribute of ‘SMTQ’ for FT-142, FT-143, FT-144, and FT-145. From Table 6 of DBD-ME-028, Attachment 19, the SMTQ designation identifies that the components require "Seismic Integrity Evaluation and Mounting of Non-safety Related Instruments Connected to ANSI 2 or 3 Fluid Systems."

Delete "Note 5" for the Variable RCP Seal Water Injection Flow on Sheet 3 of 11 in Table 7.5-7B

LDCR-SA-2012-28, EV-CR-2012-003858-2 (JDS):

FSAR section 15.5.1.2 states under section 4 of the assumptions that "The pressurizer heaters are assumed to be de-energized (the pressurizer heaters at CPNPP are automatically de-energized upon receipt of a safety injection signal by safety-related SSPS relays)." In section 15.5.1.3 of the FSAR, turning off the pressurizer heaters is incorrectly included in the discussion with SI for actions that must occur within 14 minutes of receiving an inadvertent safety injection. Corrected discussion for consistency.

LDCR-SA-2012-23, EV-CR-2012-009618-2 (CBC):

The discussion for RG 1.108 is updated to indicate the frequency for the integrated test sequence / loss of offsite power testing is revised from "At least once every 18 months" to "At the frequency specified in accordance with Technical Specification 5.5.21, 'Surveillance Frequency Control Program' ".. As documented in EV-CR-2011-002186-21, this change was evaluated and approved in accordance with TS 5.5.21.

LDCR-SA-2010-12, EV-CR-2008-003933-5 (GLM):

In the first paragraph, change direct driven open compressors to hermetically sealed type compressors.

Justification:

Plant ventilation chilled water system, subsystem 2, chiller compressors are no longer hermetically sealed type and are now direct driven open compressors. Changing from a hermetically sealed compressor to a direct drive compressor has no impact on the function of the chillers or the plant ventilation chilled water system.

LDCR-SA-2012-22, EV-CR-2010-006088-3 (JCH):

Section 3.6B.2.1.1 Reactor Coolant System (RCS) Main Loop Piping

Description: Add the following to the end of Section 3.6B.2.1.1-

"The new Leak Before Break (LBB) leak rate calculation methodology for Alloy 82/182 welds with Structural Weld Overlay (SWOL), has been approved by the Nuclear Regulatory Commission for use in Pressurizer Surge Line (PSL) LBB analysis at another nuclear power plant. This methodology is applicable for CPNPP and has been utilized for the LBB analysis for Alloy 82/182 welds with SWOL for the pressurize surge line nozzle."

Justification:

The revised LBB analysis was performed by Westinghouse and is applicable to both CPNPP U1 & U2 Pressurizer surge lines with SWOL of Alloy 52/52M. The analysis was applied at the Alloy 82/182 Weld locations. The design margin of ten on leak rate, two on flaw size and margin of one on loads remain sufficient to account for changes due to SWOL. Further, the LBBvalidation is not a safety concern since weld overlay has been applied to mitigate the Alloy 82/182 PWSCC issue. Alloy 52/52M is resistant to PWSCC. The revised analysis demonstrated that the required LBB margins are met with no adverse impact on SSCs design function as provided in the referenced UFSAR sections.

The use of LBB analysis was previously approved by the Commission. The current LBB analysis evaluates the piping with the addition of a SWOL on the DMW in order to mitigate PWSCC. The current LBB analysis uses a conservative crack morphology factor that accounts for the PWSCC on Alloy 82/182 weld. The conclusions are that all the recommended margins are satisfied. It is concluded that the previous LBB analyses conclusions remain valid and pressurizer surge line breaks should not be considered in the structural design basis of the Comanche Peak Units 1 and 2 after the SWOL application at the Alloy 82/182 weld locations.

The NRC has approved the current methodology for the Waterford Plant as documented in their safety evaluation. This NRC approved methodology may be utilized by other Licensees so it is not considered a departure from a method of evaluation under 10 CFR 50.59.

The use of the revised LBB analysis with SWOL does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design basis. Therefore NRC approval and license amendment are not required.

LDCR-SA-2012-6, EV-CR-2012-000748-2 (JDS):

Correct the characterization of the HI-TRAC to show that it is a critical load along with other new heavy loads hoisted over spent fuel in the HI-TRAC/MPC and an MPC. Other paragraph formatting and clarifications were made to reflect the use of the Holtec Dry Cask Storage. The corrected, updated and reformatted FSAR description of critical loads

LDCR-SA-2012-6, EV-CR-2012-000748-2 (JDS) (continued):

moved by the Fuel Building Overhead Crane does not change the description of the crane or its functions to control heavy loads. The correction to the description of the HI-

TRAC as being a critical load is in accordance with the CPNPP heavy loads program and NUREG-0554.

LDCR-SA-2012-16, EV-CR-2011-004569-3 (SCD):

Further detail can be found in the Engineering Basis of FDA-2011-0000133-11 and -12.

Title 10, Part 73, “Physical Protection of Plants and Materials,” Section 73.54, “Protection of Digital Computer and Communication Systems and Networks,” requires licensees provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat as described in 10 CFR Part 73, Section 73.1.

LDCR-SA-2012-26, EV-CR-2011-012069-5 (TJEW):

FSAR Table 3.9B-10 is being updated to add the active function of “Containment Penetration Protection” for FW check valves FW-0195 through FW-0202. FW check valves FW-0199 through FW-0202 are being revised to indicate that valves 1-FW-0199 through 1-FW-0202 have a Normal Position of "Closed," while valves 2-FW-0199 through 2-FW-0202 still have the Normal Position of "Open." FSAR Table 3.9B-10 is being updated and does not adversely affect or alter the design or functional requirements of the check valves.

On Table 3.9B-10, Sheet 5 of 22,

a.) for valves FW-0195, FW-0196, FW-0197, and FW-0198, add the following words in the last column after the words, "AFW Flow Path"

", Containment Penetration Protection"

b.) add the following four valves after valve FW-0198:

1-FW-0199 FW Check 6 2 Self-Acutated Closed AFW Flow Path, Containment Penetration Protection

1-FW-0200 FW Check 6 2 Self-Acutated Closed AFW Flow Path, Containment Penetration Protection

1-FW-0201 FW Check 6 2 Self-Acutated Closed AFW Flow Path, Containment Penetration Protection

1-FW-0202 FW Check 6 2 Self-Acutated Closed AFW Flow Path, Containment Penetration Protection

LDCR-SA-2012-26, EV-CR-2011-012069-5 (TJEW) (continued):

c.) Also on the same table and sheet, add "2-" to the for the followng valves:

FW-0199

FW-0200

FW-0201

FW-0202

and add the following words in the last column after the words, "AFW Flow Path"

", Containment Penetration Protection"

d.) for valves FW-0195, FW-0196, FW-0197, and FW-0198, add the following words in the last column after the words, "AFW Flow Path"

", Containment Penetration Protection"

LDCR-SA-2012-4, EV-CR-2012-000575-2 (TJEW):

Justification: Liquid mercury is known to promote liquid metal embrittlement (LME) of steels, resulting in premature intergranular cracking. There is the potential for lamp breakage to occur and mercury being released into the pools. Therefore, an evaluation by Westinghouse has been completed to provide a technical analysis of the possible effects of mercury on the integrity of the fuel, stainless steel liner plates, storage racks, and the RCS. After reviewing several possible lamp breakage scenarios and materials compatibility information and using the most limiting scenario, it was determined that the small amount of mercury contamination expected due to lamp breakage will have no significant impact on material integrity in the specified locations. The SPF filters provide removal of fission products and other contaminates by means of filtration and ion exchange. Therefore, any dissolved mercury will be removed via the purification loops.

On page 9.5-79 add the following words in the new section 9.5.3.3:

9.5.3.3 Underwater Lighting

High Pressure Sodium underwater lights are used in the spent fuel pools, transfer canals and wet cask loading pit in the Fuel Building. These lights are subject to inspections and precautions and limitations to control the free mercury contained in the bulbs. These controls minimize the likelihood of the loss of the mercury and requires an evaluation if mercury is lost and not subsequently recovered.

In Table 17A-1, sheet 41 of 58, add the following line at the end of section 37.:

LDCR-SA-2012-4, EV-CR-2012-000575-2 (TJEW) (continued):

"Spent Fuel Pool and Transfer Canal Underwater lights N/A Mfrs Stds - NONE Note C 9.5.3 Note 72, 90 "

In Table 17A-1, sheet 58 of 58, on Note 90:

Delete the word "Portable" and capitalize the "U" in "underwater"

LDCR-SA-2012-8, EV-CR-2007-002309-00-52 (JDS):

Correct descriptions of instruments located in the turbine building (i.e., non-safety related areas) and the feedwater control and bypass valve appurtenances (located in non safety related area mounted on non-safety related valves) Required corrections as noted to be consistent with other FSAR Sections (7.2.1.1.2, 8.3.1.4, and TS Bases 3.7.3). The correction that the instruments located in the turbine building and the feedwater control valve appurtenances are non-safety related is required to be consistent with FSAR Tables 17A-1 and 17A-2 which correctly show that Class 1E, seismic None items are not Nuclear Safety Related. NNS FCVs are backup to the Safety Class 2 FIVs.

These are administrative changes to correct errors made in the FSAR. The changes make these sections consistent with the other FSAR sections.

Corrected Main Feedwater Control and Bypass Valve Status. Seismic Qualification is not applicable. The correction is required to be consistent with FSAR Tables 17A-1 and 17A-2 which correctly show Class 1E, seismic None.

Deleted "safety related" with respect to the turbine trip pressure switches. Section 10.4.7 requires corrections as noted to be consistent with other FSAR Section 8.3.1.4. The correction that the instruments located in the turbine building are non-safety related is required to be consistent with FSAR Tables 17A-1 and 17A-2 which correctly show that Class 1E, seismic None items are not Nuclear Safety Related.

The change makes this section consistent with the other FSAR section.

Changes to correctly show NNS FCVs to be backup to the Safety Class 2 FIVs. Required correction as noted to be consistent with TS Bases 3.7.3.

The changes make this section consistent with the TS Bases.

Separated the Seismic None 1E devices from the Seismic 1E devices to make the application of Note 77 clearer.

LDCR-SA-2012-19, EV-CR-2012-006572-2 (TJEW):

FSAR Table 3A.3-2 references Table 3-1. References to Table 3-1 should read 3A.3-1. This is an administrative issue only and should be corrected in FSAR amendment.

LDCR-SA-2012-19, EV-CR-2012-006572-2 (TJEW) (continued):

On sheet 1 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 2 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 3 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 5 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 6 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 7 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 8 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 9 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 10 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 11 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 13 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 14 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 18 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 19 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 23 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On sheet 30 of Table 3A.3-2, change all references to Table 3-1 to Table 3A.3-1.

On page 5.3-12, last sentence of the first paragraph in section 5.3.1.6.2.1, change "Table 3-1" to "The Table below".

On page 6.2-59, last paragraph of sections 6.2.5.1.4, delete ", Table 3-1".

On page 8.3-47, second paragraph of section 8.3.1.2.4, delete the words "Table 3-1 of"

LDCR-SA-2012-24, EV-CR-2011-013842-8 (JDS):

Section 15.6.5.2.6 is revised to remove the calculated values for Peak Clad Temperature, Local Maximum Oxidation, and Core-Wide Oxidation because the values are reported in Tables 15.6-8, 15.6-9, 15.6-10, and 15.6-11. This change removes redundant information from the FSAR.

LDCR-SA-2012-24, EV-CR-2011-013842-8 (JDS) (continued):

Table is revised to include results for assessment of the thermal conductivity degradation (TCD) issue identified in NRC Information Notice 2011-21 while crediting peaking factor burndown.

LDCR-SA-2013-2, EV-CR-2011-012069-12 (TJEW):

On Table 3.9B-10, Sheet 5 of 22, for valves, FW-0195 to FW-0198, 1-FW-0199 to 1-FW-0202, and 2-FW-0199 to 2-FW-0202,

1.) replace the words under the column "Normal Position" with the words "Open/Closed" and

2.) replace the words under the column "Function (Note 5)" with "See Note 9, below"

On Table 3.9B-10, Sheet 22 of 22, add Note 9 that says,

"9) The feedwater check valves in the auxiliary feedwater lines to the steam generator open to provide the AFW flowpath during design basis events. These valves must close in the event of a line break upstream of the outboard check valve in order to terminate a Condition III loss of feedwater event. In addition, these valves also must close post-LOCA to protect the cold containment penetration during intermittent operation of AFW."

LDCR-SA-2012-18, EV-CR-2011-008432-7 (JDS):

Figures 6.2.1-1 and 6.2.1-2 are removed and the List of Figures for Section 6.0 is updated to reflect their deletion as a result of the impact of NSAL 11-5 "Westinghouse LOCA Mass and Energy Release Calculation Issues" and correct the Time for Peak Temperature for DEPSG - MAXSI.

Figure 6.2.1-1 removed as a result of the impact of NSAL 11-5 "Westinghouse LOCA Mass and Energy Release Calculation Issues" and correct the Time for Peak Temperature for DEPSG - MAXSI.

Figure 6.2.1-2 removed as a result of the impact of NSAL 11-5 "Westinghouse LOCA Mass and Energy Release Calculation Issues" and correct the Time for Peak Temperature for DEPSG - MAXSI.

Section 6.2.1.1.3.2, the reference to WPT-17251 with regards to EQ qualification and the containment pressure and temperature analysis for LOCA is removed since the EQ qualification was revised based to reflect the impact of NSAL 11-5 in RXE-LCA-CPX/0-106.

The values listed in FSAR Table 6.2.1-2 were taken from WPT-17127 "Revised LOCA Containment Response Data", Table 2. The Time for Peak Temperature for the DEPS-MAXSI was incorrectly transcribed into the FSAR from the time for peak sump

LDCR-SA-2012-18, EV-CR-2011-008432-7 (JDS) (continued):

temperature (1722.0 sec) instead of the time for peak steam temperature (24.0 sec). These values are corrected on Table 6.2.1-2.

LDCR-SA-2012-27, EV-CR-2012-010400-1 (SCD):

Administrative change from OSL (Optically Stimulated Luminescence) to TLD (Thermoluminescent Dosimeter) use at CPNPP.

LDCR-SA-2012-21, EV-CR-2012-008468-1 (SCD):

Update of titles and resumes of CPNPP Key personnel occuring since last FSAR Amendment

LDCR-SA-2012-20, EV-CR-2010-000880-7 (SCD):

Corrective actions for CR-2010-000880. ANSI-N45.2.9-1974 does not discuss electronic records, therefore the statement about duplicate storage of electronic records per the ANSI is incorrect and is being removed.

LDCR-SA-2013-3, EV-CR-2012-009307-2 (SCD):

The ability to pulse the air scrub was identified by the vendor after the system was installed. The omission of the pulsation device was overlooked and not included in the original design. Without a pulsation device the air scrub would be ineffective and this would result in permanent fouling of the Reverse Osmosis membranes.

FSAR Amendment 105a

LDCR-SA-2013-5, EV-CR-2009-005301-00-29 (TJEW):

Revise Figure 8.2-4 to show breaker E11 (CB 8075) and its air switches.

Revise Figure 8.2-5 Sheet 1 of 2 to show breaker E11 (CB 8075) and its air switches.

LDCR-SA-2013-1, EV-CR-2012-000272-18 (JDS):

The change clarifies that the top and bottom 6 inches of each fuel rod, or blanket region, typically contain fuel of lower enrichment (2.6 w/o U-235) than found in the remaining fuel rod length (central zone). This clarification is necessary because not all fuel rods have a blanket region with different enrichment. For example, the Unit 2 Cycle 14 core contains 4 fuel assemblies with blanket enrichments equivalent to the central zone enrichment of 2.308 w/o U-235.

LDCR-SA-2013-8, EV-CR-2013-001939-1 (JDS):

Updated FSAR Table 6.3-11, RWST Outflow for a Large Break LOCA with a Worst Single Failure, with the correct "Cumulative Change in RWST Volume" and correct the typo "12,46" to "12,464" Previously approved in LDCR SA-2006-010 (EVAL-2005-003364-03) but failed to be incorporated into the FSAR update. Typo is a clarifiction of volume of the RWST (12,464 gallons) or 6% indicated level of 13,450 gallons as discussed in note 11 of Table 6.3.11.

LDCR-SA-2012-12, EV-CR-2012-003165-2 (JCH):

Description: Add the following valves to Table 3.9B-10, "Active Valves"

Valve ID: DO-0111/DO-0187/DO-0211/DO-0287

System: DO

Valve Type: RELIEF

Size In.: 1-1/2

ANS Safety Class: 3

Method of Actuation: SELF-ACTUATED

Normal Position: CLOSED

Function: Pressure Relief

LDCR-SA-2012-12, EV-CR-2012-003165-2 (JCH) (continued):

Justification: These valves are required to protect diesel fuel oil transfer pumps when backpressure from the "Y" strainers requires it. Since this could happen during the Emergency Diesel Generator mission time (30 Days) and no credit had been taken for online maintenance or operator action to protect the pumps, these valves must be classifed as ACTIVE to perform their function as described in the FSAR.

LDCR-SA-2012-5, EV-CR-2012-000036-1 (JCH):

Description: The following changes to FSAR Section 10 are required to document these system changes.

1. Section 10.4.3.1.2 - Feedwater Pump Turbine Seals

Delete "The feed pump turbine seal steam system also incorporates a steam seal dump valve to establish the required flow of seal steam through the seal steam supply header."

2. 10.4.3.2.1.a Normal Operation Conditions

Delete "Excess steam pressure is dumped to the condenser through the steam seal dump valve."

3. 10.4.3.2.2.b Feedwater Pump Turbine Steam Seal System

Revise Para. 1 and 2, "In the event of excessive packing wear or blowout, the packing leakage increases drastically; this causes the steam seal supply valve to modulate close and the steam seal dump valve to open enough to maintain sealing pressure of four psig. As the SSSV controller senses a rising gland steam pressure, its pneumatic output increases to throttle close the supply valve. If the wide open steam seal dump steam seal supply valve cannot maintain pressure at four psig, the pressure increases until the relief valve starts opening at about 20 psig. At 25-psig seal header pressure, the relief valve is wide open together with the steam seal dump valve, and further increase in flow increases header pressure beyond 25 psig.

Loss of air supply to the controller causes the steam seal supply valve and the steam seal dump valve to open wide, and any pressure increase beyond 20 psig again opens the relief valve."

4. 10.4.3.5.2 Feedwater Pump Turbine Steam Seals

Revise Para. 1, "Gland steam header pressure is maintained at a three psig set point by modulation of the steam seal supply valve and the steam seal dump valve. A split range controller sets the air pressure at each the control valve so that sufficient steam is admitted from the auxiliary Steam System (if needed) and excess steam is dumped to the condenser."

LDCR-SA-2012-5, EV-CR-2012-000036-1 (JCH) (continued):

Justification:

As a result of these changes, the FWP Turbine Gland Steam System will operate with a single supply valve (u-PV-3918A) that modulates as needed to provide sealing steam and maintain header pressure at approximately 3 psig. A loss of air supply, loss of controller input or loss of controller output will cause the supply valve to fail open to provide the maximum sealing steam pressure. With the wide open supply valve, the pressure will increase until the system relief valve (uGS-0110) starts opening at about 20 psig. At 25 psig seal header pressure, the relief valve is wide open. The relief valve is sized to handle the full steam flow and pressure from a wide open supply valve. There is no dump valve to in this configuration.

LDCR-SA-2012-1, EV-CR-2010-004331-62 (JCH):

Description: Add the following to Table 9.3-9, "Failure Mode and Effects Analysis Chemical and Volume Control System Active Components - Normal Plant Operation and Load Follow" on sheet 34 of 35:

Component: 55. Motor operated gate valve 8402A

Failure Mode: Fails closed.

CVCS Operation Function: Charging and Volume Control - charging flow.

Effect on System Operation and Shutdown: Failure inhibits use of normal charging line to RCS for boration, dilution, and coolant makeup operations. Seal water injected path remains available for boration of RCS to a hot standby concentration level and makeup of coolant during operations to bring the reactor to hot standby condition.

Failure Detection Mode: Valve position indication (closed to open position chang) and group monitoring light (valve closed) at CB.

Justification:Justification:

This modification will resolve OMA Group 1 "Normal Charging Isolation" with regards to manual operator actions of motor operated valves 1(2)-8105 and 1(2)-8106. The new 3-inch motor operated valve is classified as Nuclear Safety Class 2. The control switch will be used to isolate and bypass 1(2)-HCV-0182 during maintenance and in case of a fire to ensure that the normal charging path is isolated. Replacement of this manual operated valve with a motor operated gate valve will maintain all design and operational functions of the existing valve. Addition of valve 1(2)-8402A will have similar failure modes and actions to those listed for isolation valves 1(2)-8105 and 1(2)-8106. Therefore, there is a minimal increase in the likelihood of occurrence in a malfunction.

The requirement that the valve is a "PASSIVE" open valve.

FSAR Amendment 105a

LDCR-SA-2012-1, EV-CR-2010-004331-62 (JCH) (continued):

Description: Add the following to Page 5A-10 after Normal Charging Isolatin Valves 8105 and 8106-

"Alternate Charging Isolation Valve 8402- If this normally open motor-oprerated valve closes spuriously, operator action can be used to denergized the valve operator and reopen the valve with its handwheel.

Justification:Justification:

This modification will resolve OMA Group 1 "Normal Charging Isolation" with regards to manual operator actions of motor operated valves 1(2)-8105 and 1(2)-8106. The new 3-inch motor operated valve is classified as Nuclear Safety Class 2. The control switch will be used to isolate and bypass 1(2)-HCV-0182 during maintenance and in case of a fire to ensure that the normal charging path is isolated. Replacement of this manual operated valve with a motor operated gate valve will maintain all design and operational functions of the existing valve. Addition of valve 1(2)-8402A will have similar failure modes and actions to those listed for isolation valves 1(2)-8105 and 1(2)-8106. Therefore, there is a minimal increase in the likelihood of occurrence in a malfunction.

the requirement that the valve is a "PASSIVE" open valve.

FSAR Amendment 105b

LDCR-SA-2013-15, EV-CR-2013-007598-1 (GLM):

In the 5th paragraph, clarify the description of the location of instrumentation in section 9.2.2.3.

For #7 "Thermal Barrier Cooler", clarify the description of the location of instrumentation in section 9.2.2.5.3.

Technical Justification: Correction is required by CR-2013-007598. The flow diagram (FSAR Figure 9.2-3) shows the instrumentation (temperature and flow) downstream of each thermal barrier. The ICDs attached to the CR show the redundant and diverse design. The importance of redundancy and diversity in the instrumentation and the valves should be clarified because the Thermal Barrier Rupture event is an intersystem LOCA and could challenge the CCW system outside containment. See SSER 17 for additional information.

LDCR-SA-2013-10, EV-CR-2012-012849-3 (JCH):

Section 10.2.2.7.5 Turbine Stop and Control Valves

Description: Page 10.2-12

Correct information on the closure time of the extraction steam non-return check valves from "two (2) seconds" to "one (1) second".

Justification:

The non return check valves of the extraction steam system are credited for prevention of turbine overspeed with a one second closure requirement to isolate the turbine from the moisture separator reheaters (Ref: DBD-ME-205, Rev 12, Section 4.3.1.2). The one second closure time was also specified in specification 2323-MS-021D.2,which the valve was purchased under. Changing FSAR Section 10.2.2.7.5 to identify the correct closure timerequirement will bring it into alignment with the remainder of the FSAR descriptions of the extraction steam check valves, the DBD, and the valve specification.

LDCR-SA-2013-9, EV-CR-2013-003868-1 (JDS):

FSAR 9.3.4.1.2.3, Reactor Makeup Control System, does not explicitly address the increases in RCS boron caused by burnable poison depletion during the first months of operation. Other sections of the FSAR (e.g. 4.2.2.3, 4.3.2.1, and 4.3.2.4.11) address increased in chemical shim (boron acid concentration) in the primary coolant to offset reactivity effects of burnable poison depletion.

Include description of burnable poison depletion for consistency with chapter 4 of the FSAR.

FSAR Amendment 105b

LDCR-SA-2013-20, EV-CR-2013-003797-2 (JDS):

Update FSAR 15.4.7.2 to clarify the wording related to core loading verification in a manner consistent with the referenced analysis, WCAP-16676-NP, Ref. 1 to FSAR Section 15.4.7. Clarification needed to ensure the FSAR text is consistent with the CPNPP practice and as outlined in WCAP-16676-NP.

LDCR-SA-2013-22, EV-CR-2013-008406-1 (RAS):

Revise the 2nd sentence in the last full paragraph:

"Power relief valves limit system pressure for large positive pressure transients. In the event of a large load reduction, not exceeding the design plant load rejection capability, the pressurizer power operated relief valves might be actuated for the most adverse conditions, e.g., the most negative Doppler coefficient, and the maximum incremental rod worth."

by replacing the phrase "...the most negative Doppler coefficient, and the maximum incremental rod worth." with "…Beginning of Life conditions."

Westinghouse has advised that the Analysis of Record for a large load rejection not exceeding the plant design capability for the most adverse conditions was for beginning of life, not for most negative Doppler coefficient and maximum incremental rod worth.

LDCR-SA-2013-23, EV-CR-2013-003664-1 (RAS):

Corrects an editorial error by revising a label from "HI-4" to HI-1". There is no "HI-4" signal and the indicated signal path is correctly referred to as "HI-1" in plant design documents.

LDCR-SA-2013-12, EV-CR-2013-005620-1 (TJEW):

On page 8B-2,

1.) in the third paragraph, change the word "modificaitons" to "modifications" and

2.) in the forth paragraph, remove the sentence, "In the defuelled mode, TS do not apply."

LDCR-SA-2013-19, EV-CR-2013-010298-1 (RAS):

Revise the last paragraph in section 3.6B.2.1.1 to specify use of Waterford LBB methodology for Alloy 82/182 welds with structural weld overlay.

FSAR Amendment 105b

LDCR-SA-2013-24, EV-CR-2013-008362-6 (SCD):

Transitioning Nuclear Oversight (NOS) to be a direct reporting relationship to the Vice President Station Support is based on industry experience and supports the INPO Performance Objectives and Criteria for organizational effectiveness. This organizational structure is consistent with industry practices and helps ensure that Nuclear Oversight independence remains strong. This organizational change will ensure that NOS maintains a strong reporting relationship up to the CNO and other external organizations. The responsibilities of the Director, Oversight and Regulatory Affairs have been re-assigned to the Manager, Nuclear Oversight. The personnel qualifications section, 17.2.1.5.1, has been changed to correspond with Table 13.1-1.

Correct an error that was introduced to the TMI section during incorporation of LDCR-SA 2007-001. Wording was changed to specify that the Performance Improvement Department is assigned the functions of the former ISEG organization.

This organizational change will ensure that NOS maintains a strong reporting relationship up to the CNO and other external organizations. The responsibilities of the Director, Oversight and Regulatory Affairs have been re-assigned to the Manager, Nuclear Oversight.

The responsibilities of the Director, Oversight and Regulatory Affairs have been re-assigned to the Manager, Nuclear Oversight. The personnel qualifications section, 17.2.1.5.1, has been changed to correspond with Table 13.1-1.

The responsibilities of the Director, Oversight and Regulatory Affairs have been re-assigned to the Manager, Nuclear Oversight.

LDCR-SA-2013-4, EV-CR-2012-006134-46 (GLM):

Update Section 3.5.1.4 to include the identification and description of the tornado missile impact analysis results on the Auxiliary Feedwater Pump Turbine Exhaust Stack that is not contained within reinforced concrete building or structures.

Add missing details to the FSAR for identification for this safety-related system/ components subjected to tornado-generated missiles. This also resolves the NRC question and identified NCV issue documented in the NRC Inspection Report dated 07/ 24/2012. Calculation CS-CA-0000-5493 was generated to support the impact analysis, while attempts to locate historical design information supporting the asbuilt configuration could not be located or recovered.

FSAR Amendment 105b

LDCR-SA-2013-17, EV-CR-2013-009990-1 (JDS):

The peak overpower is stated to be 118% which is inconsistent with Tables 4.1-1 and 15.0-4, which lists overpower to be 118.5%. The correct overpower was verified to be 118.5% for both units by Westinghouse LTR-TA-13-8 and LTR-TA-12-124. The overpower value has been updated to 118.5% for consistency between sections of the FSAR and with the assumptions made in the design basis analysis.

Figures 6.2.1-3 and 6.2.1-4 are both related to containment pressure temperature analysis for MSLB and do not include LOCA information. This wording which implies that these graphs contain LOCA related data was to be updated in a previous FSAR change. Reference to LOCA with regards to Figures 6.2.1-3 and 6.2.1-4 has been removed as originally required by a previous change to the FSAR (LDCR 2012-018).

A previous change to the FSAR (LDCR 2012-008) also updated the information concerning the containment analysis that supports environmental qualification evaluations in Section 6.2.1.1.3.2. This update included the phrase "modified GOTHIC run" which is misleading since no change to the actual GOTHIC code was implemented. The "modified" refers to different design inputs to GOTHIC that were used to support EQ analysis. This paragraph already describes the difference in design inputs and so "modified" has been stricken for clarity.

LDCR-SA-2013-21, EV-CR-2013-001520-1 (GLM):

In the 2nd paragraph change "...normal supply is unavailable." to "...Condensate Storage Tank is depleted."

Justification

Clarify SSW/AFW Interface to show that SSW is the long term back-up to the CST. It is used after the Tech Spec usable volume of the CST is depleted.

The Technical Specification BASES for the ST also cites the back up function. SER Sections 9.2.5, 9.2.6, and 10.4.9 are consistent with the FSAR; but are more clear that failure of the CST is not postulated. SER 9.2.5 is the most specific and makes it clear the SSI water is in addition to the CST, not in lieu of the CST.

"Additional water is also available from the service water system."

Calculation ME(B)-088 determined the flow based flow on demand after depletion of the CST rather than on full flow in lieu of the RWST. This is consistent with the design of the SSW backup function.

In the 6th paragraph update the SSI volume that would be used if the CST was unavailable in a beyond the design basis event for the current licensed thermal power of 3612 MWT

LDCR-SA-2013-21, EV-CR-2013-001520-1 (GLM) (continued):

Justification

The change administratively updates the SSI volume that would be used if the CST was unavailable in a beyond the design basis event to be consistent with the current licensed thermal power of 3612 MWT [(3612 MWT X 60 gal/MWT)/325,900 gal/acre-ft = 0.665 acre-ft].

FSAR Amendment 106

LDCR-SA-2013-7, EV-CR-2012-011569-2 (GLM):

The description of the source of potable water to the site and references to the Potable Water Storage Tank and well pumps given in FSAR Sections 9.2.4.2, 9.2.4.3, and 9.2.4.4 are changed to reflect that the site receives water solely from the Somervell County Water District and no longer uses the Potable Water Storage Tank or well pumps. The change to the source of site water is facilitated by the modifications described in FDA-2013-000044-1 and in SFER-4606965.

The modifications described in FDA-2013-000044-1 and in SFER-4606965 facilitate a change to the source of potable water to the site. As shown in 10CFR50.59 Screening 59SC-2013-000044, none of the modifications adversely affect the design bases of the Potable and Sanitary Water System described in FSAR Section 9.2.4.1. SFER-4208539 determined that the Somervell County Water District (SCWD) is capable of being the sole provider of both water pressure and capacity sufficient to fulfill the design basis functions of the Potable and Sanitary Water System. Additionally, the adequate operation of the Potable Water System with SCWD as the preferred water source since the implementation of FDA-2009-001930-01 and SFER-4208539 indicates that the Potable and Sanitary Water System can draw water solely from the Somervell County Water District and continue to fulfill its design basis functions. Therefore, all components associated with the onsite production of water can be disconnected from the rest of the system and abandoned in place.

LDCR-SA-2013-26, EV-CR-2013-001054-8 (GLM):

In the last paragraph, change the last sentence to say ".... safe shutdown of the other unit, or simultaneous shutdown of both units and maintaining them both in a safe shutdown condition."

In the 4th paragraph, change 93 acre-feet to 94 acre-feet and change 3.8 ft to 3.9 ft. In the 5th paragraph, update the max water consumption at 30 days and 39 days to 78.4 acre-feet (766'-4") and 99.4 acre-feet (765'-4"), respectively.

In the 2nd paragraph, change 765 ft 8 in to 766 ft 4 in. at 30 days and 765 ft. 4 in. at 39 days.

Technical Justification: The Safe Shutdown Impoundment Hydrothermal Analysis, ME-CA-0000-3264, was updated due to Stretch Power Uprate. It simulated one unit in a two train cooldown due to a Loss Of Coolant Accident (LOCA) and the other unit during a two train cooldown with a Loss Of Offsite Power (LOOP) as required by RG 1.27. The starting elevation is 769 feet 6 inches which is the invert of the equalization channel between the SSI and Squaw Creek Reservoir. The result for a LOCA is summarized in Table 14 of ME-CA-0000-3264 Revision 3, which shows that the maximum SSI loss due to evaporation of 93.7 ac-ft and a minimum surface water elevation after 39 days is 765 feet 7 inches. The difference between 769 feet 6 inches and 765 feet 7 inches is approximately 3.9 feet. The result for a two train cooldown with a LOOP is summarized in Table 16 of ME-CA-0000-3264 Revision 3, which shows the maximum SSI loss due to evaporation of 78.4 ac-ft at 30 days and 99.4 ac-ft at 39 days, corresponding to water

LDCR-SA-2013-26, EV-CR-2013-001054-8 (GLM) (continued):

surface elevations of 766 ft 4 in and 765 ft 4 in, respectively. Therefore, FSAR Section 2.4.11.5 need to be updated to account for both the effects of an accident in one unit, to permit the safe shutdown of the other unit, or simultaneous shutdown of both units and maintaining them both in a safe shutdown condition. FSAR Section 2.4.11.5 will also be updated to state that the minimum water elevation is 766 ft. 4. At 30 days and 765 ft. 4 in. at 39 days. FSAR Section 9.2.5.3.3 needs to be updated to state that the maximum consumption is approximately 94 ac-ft, and the decrease in surface water elevation is 3.9

LDCR-SA-2014-1, EV-CR-2013-008265-5 (JDS):

Update tables to reflect PCT errors due to ECCS model changes in Best Estimate LOCA for Unit 1 and Unit 2

LDCR-SA-2014-2, EV-CR-2012-011620-14 (SCD):

The nozzle check valve is a ASME Section III Class 1513 valve with a pressure rating of 3632 psig at 100F and a design rating of 2485 psig at 650F per drawing MD22754 Revision E (VDRT-3902424). This meets the requirements of 2323-MS-43B Category 2501 component. For flow characteristics, Westinghouse reviewed the new valve and the old valve and concluded the flow changes are insiginificant and acceptable; the Westinghouse conclusions regarding 2SI-8819A are documented in WPT-17762. The replacement valve is designed and manufactured in accordance with the requirements of ASME B&PV Code Section III, Class 1, 1974 Edition through winter 1975 Summer Addenda.

LDCR-SA-2014-3, EV-CR-2013-003295-5 (SCD):

Update table 17A-1 to include alternate penetration insert.

LDCR-SA-2014-5, EV-CR-2013-011618-2 (GLM):

In Section 1.2.2.8.2 add comma after the word "functions."

This is an editorial change only. This comma was supposed to be included via LDCR SA-2003-013, but for some reason it was not.

LDCR-SA-2014-7, EV-CR-2013-011896-1 (GLM):

Add drawing M1-0216-01 to the list of drawings for Figure 9.3-1 on page 9-xiii..

Add drawing M1-0216-01 to Figure 9.3-1.

Technical Justification: M1-0216-01 is correctly denoted as included in Figure 9.3-1; however, it is not included in the table of contents or in the FSAR Figure. This indicates the error was more than likely made when the FSAR special figures were reverted to

LDCR-SA-2014-7, EV-CR-2013-011896-1 (GLM) (continued):

plant drawings in the 1990's to reduce such errors. This is an editorial change because it corrects an error in incoporating previously approved changes.

Technical Justification: M1-0216-01 is correctly denoted as included in Figure 9.3-1; however, it is not included in the table of contents or in the FSAR Figure. This indicates the error was more than likely made when the FSAR special figures were reverted to plant drawings in the 1990's to reduce such errors. This is an editorial change because it corrects an error in incoporating previously approved changes.

LDCR-SA-2014-6, EV-CR-2013-010811-1 (SCD):

The gamma detectors provide the radiation monitoring during routine operation at CPNPP while the additional beta/gamma radiation detectors availability are necessary for less frequent and emergency condition which might warrant them.

LDCR-SA-2014-9, EV-CR-2013-008403-6 (SCD):

Incorporated editorial corrections to the last paragraph in section 6.3.5.4.1 by replacing "taking" with "the" in the first sentence and "less" with "loss" in the second sentence.

“The change from 28 ft to 22 ft for required NPSH is based on the vendor. The change in NPSHa from 32 ft (based on a previous empty alarm setpoint) to 26.7 ft is based on the current alarm setpoint (based on the end of injection mode of operation as described in Section 6.3.2.2.10.2). These changes reflect corrections to the NPSH calculations to reflect the current plant design.”

LDCR-SA-2014-11, EV-CR-2013-008397-4 (TJEW):

The following is an NRC finding documented in the NRC Component Design Bases Inspection Report, IR 2013007:

The inspectors identified a Severity level IV, non-cited violation of 10 CFR 50.71(e)(4), requires the UFSAR be updated, at intervals not exceeding 24 months, and states in part, "the revisions must reflect all changes made in the facility or procedures described in the UFSAR." Specifically, prior to June 20, 2013, the inspectors identified the alternate power diesel generator system was not described in sufficient detail in the FSAR as required. This LDCR corrects the problem.

On page 8.3-1, Section 8.3.1.1. subsection 3, add the following words, "CPNPP has also provided a set of Alternate Power Diesel Generators (APDGs) for each unit with the capability to connect to a Class 1E train at a time to provide defense-in-depth for safe shutdown of a unit during outages or during extended duration of an inoperable offsite circuit on occurrence of a beyond design basis event of concurrent loss of offsite power and failure of EDGs. The APDGs may provide 3450 kVA to provide long term cooling of each unit to respond to a beyond design basis event."

FSAR Amendment 106

LDCR-SA-2012-13, EV-CR-2012-005259-3 (TJEW):

Figure 1.2-1 is revised to show the location of XST1A

Page 8.2-1 will add a new paragraph after the 7th paragraph and will state, "Startup transformer XST1 and alternate startup transformer XST1A are connected to a common overhead line from the 138-kV switchyard. Each transformer is provided with a 138-kV motor-operated air switch such that each transformer can be energized independent of the other transformer."

In the last paragraph on the same page, the word "spare" will be replaced with "alternate"

On page 8.2-2, after the first partial paragraph, the following new paragraph will be added, "Alternate startup transformer XST1A is located under the 138-kV line to XST1 (refer to Figure 8.2-1) to serve as a replacement of XST1 after a future plant modification to connect cable buses from secondary X and Y windings of XST1 and XST1A to transfer panels to provide 138-kV offsite power to Units 1 and 2 safety related buses."

In the next paragraph after the new paragraph above, replace the word "Spare" with "Alternate", delete the word "dedicated" and deleted the last sentences that says, "This spare transformer may be physically relocated to a dedicated location near XST1, to serve as a replacement of XST1."

Figure F 8.2-1 is revised to show location of alternate startup transformer XST1A and revise designation of XST2A.

On Figure 8.2-4 show the electrical network interconnection of XST1/ XST1A and revise the detail for XST2/XST2A.

On Figure 8.2-7 Sheet 1, XST1A willl be added to the transmission lines going to XST1 from the 138kV switchyard.

On Figure 8.2-7 Sheet 2, XST1A will be added to the 138kV transmission tower.

Figure 8.2-9 will be updated to show transformer XST1A in the 138kV transformer connections plan and elevations and new air switch configuration for transformers XST1 and XST1A

Figure 8.2-11 will be updated to reflect the as-built cable bus configuration.

New Figure 8.2-11A will show the as built details of the XST1A cable bus configuration to the 6.9kV switchgear.

On page 9.5-17, last sentence of section 9.5.1.5.6, replace the word "spare" with "alternate, and at the end of the paragraph add the following words, "Alternate transformer XST1A is separated from adjacent structures by a three-hour rated fire wall."

FSAR Amendment 106

LDCR-SA-2014-15, EV-CR-2013-008397-12 (TJEW):

The linkage to beyond design basis event is being deleted as a result of LDCR SA-2014-011 because the FSAR is a design basis document, i.e., a 10CFR50.2 plant, in FSAR section 8.3.1.1.1.

On page 8.3-1, LDCR SA-2014-011 inserted a paragrapgh after section 8.3.1.1.1.3 that said, "CPNPP has also provided a set of Alternate Power Diesel Generators (APDGs) for each unit with the capability to connect to a Class 1E train at a time to provide defense-in-depth for safe shutdown of a unit during outages or during extended duration of an inoperable offsite circuit on occurrence of a beyond design basis event of concurrent loss of offsite power and failure of EDGs. The APDGs may provide 3450 kVA to provide long term cooling of each unit to respond to a beyond design basis event."

This LDCR will modify the above paragraph as follows:

1. In the first sentence, add the words, "non-safety related" before Alternate Power Diesel Generators.

2. In the first sentence delete the words, "a beyond design basis event of."

3. In the second sentence delete the words, "to respond to a beyond design basis event."

4. Add the following sentence to the end of the paragraph, "See Section 8.3.1.2.1.7.h for a discussion of the sizing of APDG cables."

On page 8.3-38,

1. second paragraph and second sentence of section h., relocate the words "during modes 5 and 6" to after the word "APDGs."

2. third paragraph and first sentence of section h, replace the "," between 6,9kV to a "period."

LDCR-SA-2014-14, EV-CR-2013-008362-17 (SCD):

Due to CPNPP organizational and upper management changes, Chapter 17 and the TMI Section of the FSAR is being revised.

LDCR-SA-2014-16, EV-CR-2013-011292-1 (JDS):

Update core thermal power output from 3458 MWt to the currenctly licensed power level of 3612 MWt.

FSAR Amendment 106

LDCR-SA-2014-17, EV-CR-2014-001610-1 (JDS):

Include exception for use of the 193 core offload spaces for storage to assure ability to comply with Cask Technical Specifications. This will assure that 32 spaces will remain available for cask offload.

Clarification. The word "hose" should be "hoist".

LDCR-SA-2014-18, EV-CR-2012-005259-93 (TJEW):

Add Figure 8.2-11A to the List of Figures for Chapter 8 of the FSAR on page 8-iv:

8.2-11A Spare Start-up Transformer XST1A Cable Bus Connection to 6.9 KV Switchgear

LDCR-SA-2014-13, EV-CR-2014-006885-1 (SCD):

Due to CPNPP organizational and upper management changes, Chapter 13 of the FSAR is being revised.

LDCR-SA-2011-4, EV-CR-2010-005675-1 (JDS):

Update UFSAR Section 15.6.2 to provide revised post-SPU EAB and LPZ estimated doses for a CVCS letdown break during a system alignment using 3 letdown orifices at Mode 3 operation where the system is below 500 degrees F. The Thyroid and Whole Body doses limits for the EAB and LPZ are based on a small fraction (10%) of the 10CFR100 as directed by the applicable SRP guidelines. 10CFR100 dose limits for the thyroid and whole body doses are 300 rem and 25 rem, respectively. The resulting dose margins derived remain less than a small fraction of the 10CFR100 limits.

FSAR Amendment 106a

LDCR-SA-2014-20, EV-CR-2013-003649-5 (JCH):

Table 3.9B-10: Active Valves

Description: Correct typo for valve 2DO-0052 (shown as 2DD-0052 on Sheet 15)

Move valves DO-0111, DO-0187, DO-0211 on page 19 and DO-0287 on page 20 to sheet 15 and change the function of these valves from Pressure Relief to Recirculation Flow Path.

Justification: Pressure relief is not the active safety function of the subject valves. These valves provide Recirculation Flow Path (miniflow) for the fuel oil pumps. The subject valves were also inserted in the wrong location in the table.

LDCR-SA-2013-6, EV-CR-2010-009018-22 (JDS):

Clarification by providing a more detailed nominal dmension of the Region 1 spent fuel storage racks in the fuel building.

Update the minimum Boron Concentration for the spent fuel pools to be consistent with License Amendment 162.

Replace Table 9.1-4 with a revised table reflecting the revised spent fuel pool criticality analysis approved in TS amendment 162.

Update criticality codes used to reference SCALE 5.1 and PARAGON to reflect codes used in the spent fuel pool criticality analysis.

Update to include descriptive information for the revised spent fuel pool criticality analysis approved in amendment 162 and revise the references to the WCAP used in the license amendment.

LDCR-SA-2014-21, EV-CR-2014-010150-2 (SCD):

This activity lowers the minimum acceptable filter mesh size of the Reactor Coolant System (RCS) Filter from 0.1 micron to 0.05 micron in DBD-ME-255 to reduce source term dose in the Chemical Volume and Control System (CVCS).

LDCR-SA-2014-26, EV-CR-2013-006623-5 (JDS):

Correct inconsistency between Table 6.2.4-1,Table 6.2.4-2, Figure 6.2.4-1 and Table 6.2.4-3.

Update the function and classification of 1/2-HV-2154 and 1/2-HV-2155 to no longer function as Containment Isolation Valves. The containment isolation function was previously transferred to locked closed manual isolation valves, FW-0113 and FW-0116. FSAR Tables 6.2.4-1 and 6.2.4-2 identify FW-0113 (Item #22) and FW-0116 (Item #20) as the Feedwater Sample line containment isolation valve for penetration MI-5 and MI-6,

LDCR-SA-2014-26, EV-CR-2013-006623-5 (JDS) (continued):

respectively. Table 6.2.4-1 references Figure 6.2.4-1 Arrangement 16 as the containment isolation valve arrangement which depicts a locked closed isolation valve outside of containment. Table 6.2.4-1 also references FSAR Figure 10.4-9 which shows FW-0113/0116 as normally locked closed for containment isolation. Table 6.2.4-2 describes the isolation valves as locally operated manual valves. This is consistent with DBD-ME-203, Rev. 36 Section 4.3.1.2.i and 5.4.1 which identify a locked closed manual valve containment isolation arrangement fulfilled by FW-0113 and FW-0116. However, previous FSAR updates missed the removal of the reference to the properties of the former containment isolation valves in Table 6.2.4-3 and incorrectly describes the valves as having a containment isolation signal, power source, and closure time. The Table also incorrectly shows the normal valve position as open. DBD-ME-013, Attachment 2 (Items 20 and 22) provides the correct properties for FW-0113/0116 that should be reflected in Table 6.2.4-3.

Table 6.2.4-3 is updated to be consistent with DBD-ME-203, DBE-ME-013, FSAR Tables 6.2.4-1 and 6.2.4-2, and FSAR Figure 6.2.4-1.

LDCR-SA-2014-12, EV-CR-2014-005546-1 (SCD):

Spent fuel pool swing gate pressure seals have been re-classified as safety related from not safety related components. Additional information will be added to show the new spent fuel pool swing gate equipment as shown in the following attachments. (Reference FDA-2013-000052-01)

LDCR-SA-2014-22, EV-CR-2014-005228-1 (SCD):

FSAR Tables 3.2-3 and 3.2-4 are updated to reflect the flow diagram split that created drawings M1-0269 Sh. C and D.

The Ch. 11 Listing of Figures is updated to reflect the flow diagram split that created drawings M1-0269 Sh. C and D.

Figure 11.3.-1 is updated to reflect the flow diagram split that created drawings M1-0269 Sh. C and D.

LDCR-SA-2014-19, EV-CR-2013-008947-12 (SCD):

The pressure gauges for the airlocks are special cases due to their Safety Class 2 function as part of the airlocks. Typical pressure gauges on Safety Class piping are classified as NNS because they are isolable and/or minor leakage is acceptable. The subject valves are not accessible and post-LOCA increase in leakage is not acceptable. Therefore, they are being classified as ANSI Safety Class 2, Seismic Category I under FDA-2012-000230-02.

FSAR Amendment 106a

LDCR-SA-2014-25, EV-CR-2014-002884-11 (JDS):

Revise Note (a) to Table 15.0-4 (applicable to Overtemperature N-16 function). The response times of RCS Cold Leg RTD and OTN16 RTD instrumentation is changed from 6.0 and 3.3 seconds to 6.7 and 3.3 seconds respectively. However, this does not change the overall OTN16 RTD response time requirement of 9.3 seconds and therefore has no adverse affect on the operation of the Tcold RTD signal or the existing safety analyses for the OTN16 function since the safety analyses is insensitive to a trade-off between the RTD response time and the instrumentation response time. Therefore, there is no change to overall response time of 9.3 seconds assumed in the applicable safety analyses.

LDCR-SA-2015-1, EV-CR-2015-000856-1 (SCD):

Additional information will be added to show the new spent fuel pool swing gate equipment. LDCR SA-2014-012 added multiple changes to table 17A-1 and it was found that the addition to system 16 was incorrect and needs to be removed.

Paragraph added to section to describe the passive nature of the refueling gate seals.

FSAR Amendment 106b

LDCR-SA-2015-4, EV-CR-2011-013548-1 (JEB):

ISFSI Security Plan and Cyber Security Plan have been added to the CPNPP security program. Update FSAR to include the new plans by simplifying the section 13.6 statement to make it inclusive of all the plans called for in 10CFR73:

Proposed Wording:

A comprehensive security program has been developed at CPNPP in accordance with the requirements of applicable portions of 10CFR73, "Physical Protection of Plants and Material." The security plans that describe the program contain security related information (SRI) or safeguards information (SGI) and are withheld from public disclosure in accordance with 10CFR2.390 or 10CFR73.21. The CPNPP security plans are presented in separate submittals.

LDCR-SA-2014-23, EV-CR-2010-004331-98 (RAS):

Add the following description for the AFW valves: "The three AFW valve are normally de-energized to prevent a fire induced hot-short from causing a mal-operation of the valve(s). This also de-energized position indication in the Control Room. Therefore, these valves are "Locked Closed" to provide positive indication of correct valve position during normal operation."

LDCR-SA-2015-7, EV-CR-2014-012820-1 (SCD):

This LDCR is a clarification of information in Section 6.5.1.1, which states "Efficiencies of the HEPA filters and iodine adsorbers are in accordance with Table 9.4-4." Testing of the HEPA filters following significant painting, a fire or chemical hazard ensure compliance with the efficiencies in Table 9.4-4. FSAR Table 6.5-1 is revised where Regulatory Guide 1.52, Revision 2 is used by CPNPP rather than Revision 1.

LDCR-SA-2014-28, EV-CR-2014-005590-3 (JDS):

The results obtained from ANSYS version 14.0 for 24 of the 26 problems compared were found to have exact numerical matches to those obtained from version 13.0. The results from the 2 problems that did not match were further evaluated for the significance of their mismatches. The detailed evaluation of these 2 problems found that the results from one differed by less than 1-percent which is the ANSYS expected analytical tolerance. A reviewof the input script for the second problem discovered that the version 14.0 input script was different than what had been used in version 13.0. When the script used for version 13.0 was evaluated by ANSYS version 14.0, it was found to differ by less than 1-percent which is the expected ANSYS analytical tolerance. The input script is user provided input that describes the analytical problem to be solved by the program. A change to the input script is not a change to the program. A review of changes made by ANSYS did not find that any significant issues or problems had been corrected in version 14.0. It is therefore the conclusion of this Evaluation that the analysis results obtained from ANSYS version 14.0 are essentially the same as those obtained from ANSYS

LDCR-SA-2014-28, EV-CR-2014-005590-3 (JDS) (continued):

version 13.0 and ANSYS version 14.0 is not considered a departure from a method of evaluation described in the UFSAR.

LDCR-SA-2015-8, EV-CR-2015-000067-6 (JDS):

FSAR section 9.1.3.2 currently states that the SFP Boron concentration is "approximately the same" as the RWST. As described in EV-CR-2015-000067-1, it is desired to maintain the Spent Fuel Pool boron value between 2600-3000 ppm. The RWST is normally maintained 2400-2600 ppm, which is the TS limit applicable in MODES 1-4 of operation, during which SFP and RWST water do not mix. In Emergency Operating procedures, guidance exists to fill the RWST from the SFP during beyond-design-basis events. As described in EV-CR-2015-000067-1, the boron concentration of this emergency water source does not need to satisfy the normal RWST TS limits, but has its own Technical Specification requirement and SFP water at a boron concentration of up to 3000 ppm remains a valid source of water during this potential beyond-design-basis event.

LDCR-SA-2015-10, EV-CR-2012-011869-1 (JDS):

Revise Section 6.2.2.2 to reference a single table 6.2.1-9, as Tables 6.2.1-9A and 6.2.1-9B have been combined into a single table in Amendment 103 (LDCR-2008-013).

Revise 6.3.2.8 to remove obsolete information (references to Appendix 4A and 4B which were previously removed in LDCR-SA-2008-013) and provide an appropriate reference to Tables 6.2.1-9 and 6.2.1-12.

ReviseTable. 6.3-7 to remove obsolete information (references to Appendix 4A and 4B which were previously removed in LDCR-SA-2008-013) and provide an appropriate reference to Tables 6.2.1-9 and 6.2.1-12.

Delete the last sentence on the page since Cycle/Unit specific information for Units 1 and 2 is no longer provided in Appendices 4A and 4B as these Appendeces were previously removed in LDCR-SA-2008-013.

FSAR Amendment 107

LDCR-SA-2015-12, EV-CR-2015-006885-2 (SCD):

6.3
In the PSAR (RESAR 3), passive failures were considered coincident with hot leg recirculation at 24 hours. The FSAR originally assumed hot leg recirculation at 24 hours; however, that has changed over time without addressing if the passive failure timing. FDA-2015-000095-01 clarifies the DBD based on the conclusion that the passive failure was considered at switchover to hot leg recirculation and that would be the current licensing basis. Therefore, the FSAR should be updated in accordance with 10CFR50.71(e).

LDCR-SA-2015-13, EV-CR-2015-002140-2 (GLM):

9.2-2
In the last paragraph, correct the CCW fouling factor from 0.0017 to 0.0020.

Justification
Errors were made in previous LDCRs (LDCR-SA-2001-021 and LDCR-SA-2010-001) as described in AI-CR-2015-002140-1. LDCR-SA-2001-021 defined the design fouling as 0.0020 with no plugging and 0.0015 with the plugging allowance. Plugging allowances are no longer used nor are they required. LDCR-SA-2010-001 changed the design fouling factor in two places (not all) and did not change the factor with plugging. The 10CFR50.59 basis for the 2010 LDCR was FDA-2006-003080-01 and -02. These FDAs discuss fouling; HOWEVER, they did discuss a change in the design fouling factor. The actual design fouling factor was not changed. Calculation RXE-LA-CPX/0-018 R8 dated 10/28/08 was performed at the design fouling of 0.002. Calc RXE-LA-CPX/0-018 R9 for power uprate did not change the design fouling; however, it noted it would exceed the 135F limit on the operating unit during a two train CCW outage on an outage unit. Per the attached email and RXE-LA-CPX/0-018 R9, the current design fouling remains 0.002. RXE-LA-CPX/0-020 Rev. 9 for power uprate used a fouling factor of 0.0017 for the single train forced cooldown. The reduction was made to maintain the normal operating temperature limit of 122 F used for the piping analysis. The subject cooldown would be an abnormal condition no subject to the 122F limit. The impact review CVC for this calc revision was not in conformance with ECE-5.03 or ECE-5.01. This revision was the basis for LDCR-SA-2010-001. This calculation was subsequently revised to make corrections. The current version of RXE-LA-CPX/0-020 is R12 dated 11/12/2013 and the design fouling of 0.0020 has been restored. STA-734, Service Water System Fouling Monitoring Program, includes the actual number of plugged tubes in the monitoring based on calculation ME-CA-0229-2188. Plugging limits are not used.

Therefore, correction of the design fouling factor and the deletion of the plugging allowance are required to update the FSAR to be current as required by 10CFR50.71€.

9.2-3
In the first paragraph, correct the CCW fouling factor from 0.0017 to 0.0020. Delete the first and third sentences. In the last sentence, insert the word "fouling" between "periodic" and "monitoring."

LDCR-SA-2015-13, EV-CR-2015-002140-2 (GLM) (continued):

Justification
Errors were made in previous LDCRs (LDCR-SA-2001-021 and LDCR-SA-2010-001) as described in AI-CR-2015-002140-1. LDCR-SA-2001-021 defined the design fouling as 0.0020 with no plugging and 0.0015 with the plugging allowance. Plugging allowances are no longer used nor are they required. LDCR-SA-2010-001 changed the design fouling factor in two places (not all) and did not change the factor with plugging. The 10CFR50.59 basis for the 2010 LDCR was FDA-2006-003080-01 and -02. These FDAs discuss fouling; HOWEVER, they did discuss a change in the design fouling factor. The actual design fouling factor was not changed. Calculation RXE-LA-CPX/0-018 R8 dated 10/28/08 was performed at the design fouling of 0.002. Calc RXE-LA-CPX/0-018 R9 for power uprate did not change the design fouling; however, it noted it would exceed the 135F limit on the operating unit during a two train CCW outage on an outage unit. Per the attached email and RXE-LA-CPX/0-018 R9, the current design fouling remains 0.002. RXE-LA-CPX/0-020 Rev. 9 for power uprate used a fouling factor of 0.0017 for the single train forced cooldown. The reduction was made to maintain the normal operating temperature limit of 122 F used for the piping analysis. The subject cooldown would be an abnormal condition no subject to the 122F limit. The 9.2-3 impact review CVC for this calc revision was not in conformance with ECE-5.03 or ECE-5.01. This revision was the basis for LDCR-SA-2010-001. This calculation was subsequently revised to make corrections. The current version of RXE-LA-CPX/0-020 is R12 dated 11/12/2013 and the design fouling of 0.0020 has been restored. STA-734, Service Water System Fouling Monitoring Program, includes the actual number of plugged tubes in the monitoring based on calculation ME-CA-0229-2188. Plugging limits are not used.

Therefore, correction of the design fouling factor and the deletion of the plugging allowance are required to update the FSAR to be current as required by 10CFR50.71€.

LDCR-SA-2015-15, EV-CR-2015-007540-5 (GLM):

9.2-3
After the last sentence in Section 9.2.1.2.2, add clarification that more restrictive limits are imposed on SSI temp and CCW Hx fouling on the operating unit during outages.

Technical Justification: CCW design fouling and fouling monitoring are described in the FSAR for all normal and accident conditions. However, during the abnormal condition where the CCW on an outage unit is unable to provided cooling to common shared loads, the operating unit is capable of cooling the essential loads. However, restrictions on valve alignments, SSI temperature and/or CCW heat exchanger fouling may be imposed to assure operability is not affected. It may be neccesary to limit the number of supplied components to protect the CCW pumps. CR-2014-007235 is an example of such precautions taken. FDA-2014-000175-01 clarified DBD-ME-229 for the alignment of both spent fuel pool heat exchangers to one unit.

FSAR Amendment 107

LDCR-SA-2015-15, EV-CR-2015-007540-5 (GLM) (continued):

9.2-7
In the last sentence on the page, add the word "normally" between "is" and "required." After the last sentence on the page, add clarification that during outages it may be necessary to take both safeguards trains and/or the non-safeguards loop of CCW out of service for maintenance. In this case, the common components are cooled by the operating unit. In this abnormal alignment, more restrictive limits may be imposed as described in Section 9.2.1.2.1.

Technical Justification: CCW design fouling and fouling monitoring are described in the FSAR for all normal and accident conditions. However, during the abnormal condition where the CCW on an outage unit is unable to provided cooling to common shared loads, the operating unit is capable of cooling the essential loads. However, restrictions on valve alignments, SSI temperature and/or CCW heat exchanger fouling may be imposed to assure operability is not affected. It may be neccesary to limit the number of supplied components to protect the CCW pumps. CR-2014-007235 is an example of such precautions taken. FDA-2014-000175-01 clarified DBD-ME-229 for the alignment of both spent fuel pool heat exchangers to one unit.

LDCR-SA-2015-18, EV-CR-2014-013507-2 (SCD):

In the FSAR Table 6.3-3 Sheet 2 of 3, in the "Valve Identification" column for the Charging pump miniflow isolation valves it lists the valve numbers as "8810 and 8111". This is incorrrect, it should be "8110 and 8111".

Technical Justification:
Administrative change to correct error in valve identification.

LDCR-SA-2015-16, EV-CR-2015-008796-1 (SCD):

Clarify QA review of procedures in SORC.This is to support removal of ODA's and MDA's from SORC approval IAW EV-CR-2013-001259-1.

LDCR-SA-2015-17, EV-CR-2015-000973-2 (GLM):

Description of Change: Delete reference to Figure 9.3-3 on List Of Figures (Page 9-xiv).

Technical Justification: Flow diagrams M1-0216-01 and M2-0216-B were changed from referencing FSAR Figure 9.3-3 to referencing FSAR Figure 9.3-1 in CR 2014-008461. This change updates the List Of Figures to reflect this.

FSAR Amendment 107

LDCR-SA-2014-4, EV-CR-2012-0002652-27 (CBC):

The FSAR is updated to reflect changes to the plant and plant site required by NRC Orders EA-12-049 and EA-12-051. These changes are for “Beyond-Design-Basis External Events” and are not used nor credited by the current plant design basis. Section 1.2 is updated to reflect a general description of NRC Orders EA-12-049 and EA-12-051. Figure 1.2-1 is updated to show the site contains a FLEX Storage Building. The FLEX Storage Building protects equipment credited for NRC Order EA-12-049. Sections 9.1.3.2 (System Description) and 9.1.3.5 (Instrument Requirements) are updated to reflect the addition of the Spent Fuel Pool Level Instrumentation. FSAR Tables 17A-1 and 17A-2 are updated to reflect the quality assurance attributes of the Spent Fuel Pool Level Instrumentation.

LDCR-SA-2014-27, EV-CR-2014-012821-1 (JDS):

Section 6.2-3: Add a sentence that reflects that the temperature and pressure profiles utilized for EQ evaluations as described in section 3.11B bounds the limiting temperature and pressure transients for postulated LOCA and MSLB events. The sentence describes the relationship between containment integrity and environmental qualification. This addition ensures that changes to the containment integrity analysis will consistently cascade to the EQ analysis.

FSAR Table 6.2.1-2 describes CPNPP’s compliance with GDC-16 and -50 that the peak calculated containment pressure should be less than the containment design pressure of 50 psig following a LOCA and is based on the double-ended hot leg (DEHL) and double-ended pump suction leg (DEPS) breaks assuming minimum and maximum ECCS performance (MINSI and MAXSI) cases analyzed by Westinghouse using the EPITOME code. NSAL 14-2 identified errors in the LOCA M&E release analyses input modification program database and the input preprocessor. The analyses are sensitive to energy stored in the RCS metal, including the SG tubes and the program was using the density for stainless steel in determining the mass of the SG tubes and the stored energy based on a stainless steel specific heat. CPNPP steam generators are Alloy 690 (U1) and Alloy 600 (U2) which have different specific heat values than stainless steel. RXE-LA-CPX/0-106 “Impact of M&E Analysis Errors on CPNPP Post LOCA Containment Pressures and Temperatures Meeting Acceptance Criteria” determined that the pressure penalty is 0.2 psi and the temperature penalty is 0.3°F. These penalties were applied to the peak pressure and temperature values in the attached markup of Table 6.2.1-2.

The note added to Tables 6.2.1-2 and 6.2.1-2A as well as Section 6.2.1.1.3.2 describes the relationship between containment integrity and environmental qualification. This addition ensures that changes to the containment integrity analysis will consistently cascade to the EQ analysis.

FSAR Amendment 107

LDCR-SA-2015-19, EV-CR-2015-011212-1 (SCD):

Changes to CPNPP organizational structure due to management changes and restructuring.